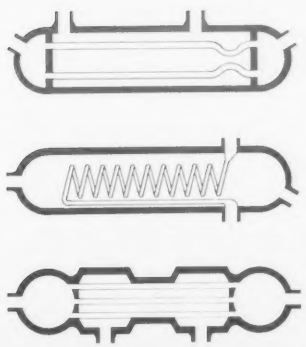
POWER REACTOR TECHNOLOGY

A Quarterly Technical Progress Review

Prepared for DIVISION OF TECHNICAL INFORMATION, USAEC, by
ARGONNE NATIONAL LABORATORY



Spring 1966

● VOLUME 9

● NUMBER 2

Liquid-metal-heated steam generator...see p. 69

TECHNICAL PROGRESS REVIEWS

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Contents

Power Reactor Technology

Vol. 9, No. 2

Issued quarterly by the U. S. Atomic Energy Commission. Use of funds for printing this publication approved by the Director of the Bureau of the Budget on October 14, 1963.

Power Reactor Technology is a quarterly review of developments in reactor technology—including new concepts and applications as well as research results, analysis, and experience with existing or new components and systems. Prepared by Argonne National Laboratory at the request of the Division of Technical Information of the U. S. Atomic Energy Commission, *Power Reactor Technology* is intended for reactor designers and other technical specialists concerned with the interplay among reactor development, design, construction, operation, and economics. Thus this quarterly journal reports and interprets progress in the power-reactor field in terms of its significance to the reactor designer. Its scope includes research and development in the various technical specialties in the field as well as design and construction practices, reactor concepts, applications, economics, and operating experience.

Some articles summarize and critically evaluate reported developments; others review a topic broadly so that designers can interpret new developments, experience, and trends. All, however, call the reader's attention to reports and publications that merit study. Because any appraisal involves the reviewer's opinion of the significance of the work reported, readers are urged to consult the references for additional information and the judgments of the original authors. If a reader has information that causes his evaluation to reinforce, modify, or contradict the opinions of our reviewers, he is encouraged to write to the Editor.

For reasonably timely coverage, articles often must be based on fragmentary information. Thus, whenever appropriate, continuity will be maintained from one issue to another by use of subject headings that reflect parts of the following scope of the journal:

> CONCEPTS and APPLICATIONS: Progress in evaluating the applicability and economics of various reactor types and systems (including unconventional approaches), as well as of fuel resources and cycles, for utility central-station generation of electricity, auxiliary power, process radiation and heat, desalting, and propulsion—and for terrestrial, undersea, aerospace, and other advanced uses.
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> ANALYSIS and EXPERIMENTATION: Advancements in the techniques of reactor physics, fluid and thermal technology, power conversion, fuel elements, materials, mechanics, control and dynamics, reactor and radiation safety, containment, and siting.
>
> SYSTEMS and COMPONENTS: Experience as reflected in design and construction practice, components, systems technology, and operating experience of various specific types of reactors—including pressurized- and boiling-water reactors, molten-salt, organic-, gas-, and liquid-metal-cooled reactors, as well as generally applicable aspects of research and test reactors.

At Argonne National Laboratory, the Reactor Engineering Division (L. J. Koch, Director) has the principal responsibility for the preparation of *Power Reactor Technology*, with the regular assistance of the Reactor Physics Division (R. A. Avery, Director) and the occasional assistance of other Laboratory divisions (Metallurgy, Chemical Engineering, Chemistry, Reactor Operations, and Idaho). Thus, unless noted otherwise, the reviewers and authors of articles are Laboratory staff members. The editors will welcome interpretive review articles contributed by readers outside the Laboratory.

Section

I

Power Reactor Technology

Fluid and Thermal Technology

Compressibility Affects Flow Instability and Burnout

By Hans K. Fauske

The problem of evaluating accurately the burnout heat flux (maximum or critical) and flow instabilities in forced-convection boiling systems has received a great deal of attention in the last 10 years, primarily because of the need for better correlations in nuclear reactor design. However, the results obtained to date form a somewhat confused picture; there are numerous widely differing correlations and contradictory conclusions. This is largely because unstable behavior and "premature" burnout can be caused by a number of mechanisms that are not always properly accounted for. This article deals with the importance of including compressibility effects when two-phase flow data are being interpreted.

Compressibility and Flow Oscillation

Of particular concern has been the problem of stable vs. unstable or premature burnout heat flux caused by flow oscillations in a boiling channel in a reactor. The presence of compressibility, either within or upstream of the test section, represents an energy-storage mechanism that is capable of producing large-amplitude oscillations from smaller ones; the smaller ones are usually present in two-phase flow systems.

The introduction of upstream compressibility, in the form of an accumulator filled with compressible medium, results in pulsating flow conditions and severe reductions in the obtained critical heat fluxes.[1,2] Also, large oscillations resulting from two-phase compressibility within the boiling channel have been observed.[3,4]

The instability can be removed by increasing the available pressure drop and installing a throttle valve near the boiling channel to effectively steepen the pump characteristic. The amount of throttling necessary to obtain stable burnout data in a single boiling channel has been investigated.[1] A study of similar nature[5] shows that the compressibility caused by a volume of subcooled water upstream of the heated test section was sufficient to cause thermal instability; large throttling was necessary to stabilize the flow. A recent comprehensive study deals with instability caused by both flow excursion and flow oscillations; the onset of instability agrees well with the theoretical predictions.[6] The analytical results were verified on a single-tube apparatus with controlled compressibility.

Compressibility and Choking

The increased tendency for large flow oscillations, which eventually reduce the maximum obtainable heat flux, is not the only detrimental effect caused by the presence of compressibility. Another problem is that of two-phase choking (maximum, critical, or sonic flow), which, like single-phase choking, can result in additional instability problems. Two-phase choking flow becomes particularly important in boiling systems operating at relatively low pressures and high inlet superficial liquid velocities. However, the literature reveals that little or no attention has been given to this aspect of compressibility and to its effect on stable-flow conditions and burnout heat fluxes.

Data Interpretation

The difficulties in interpreting experimental data that result from not checking for sonic

effects are illustrated by the data obtained by Lowdermilk et al.[1] Because Ref. 1 is the only available source of low-pressure burnout data for the net boiling regime, it often is cited to illustrate how flow oscillations affect the local heat-transfer characteristics and the critical heat flux.[7] However, the use of these data in investigations[6,8] in which the possibility of choking was overlooked has led to misinterpretations. Correlation of such data on the basis of local conditions is not possible because in Ref. 1 exit quality was based on the pressure in the downstream plenum chamber. For most of the stable runs, it can be shown that the pressure at the tube exit must have been considerably higher because of the occurrence of choking (as first mentioned by this writer in an oral discussion on critical flow held at the 1964 National Heat Transfer Meeting, Cleveland, Ohio).

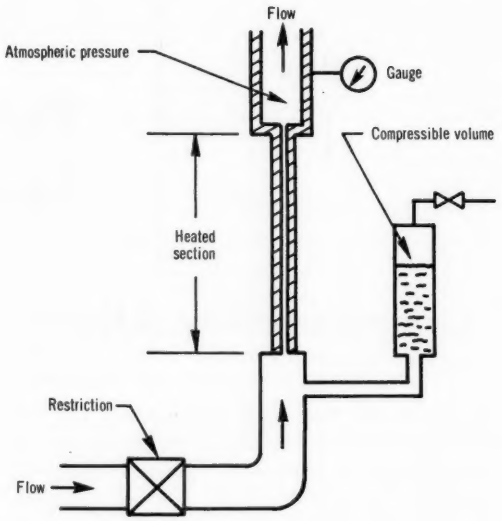

Fig. 1 Test apparatus used by Lowdermilk et al.[1] provided for measuring pressure in downstream plenum rather than at test-section exit.

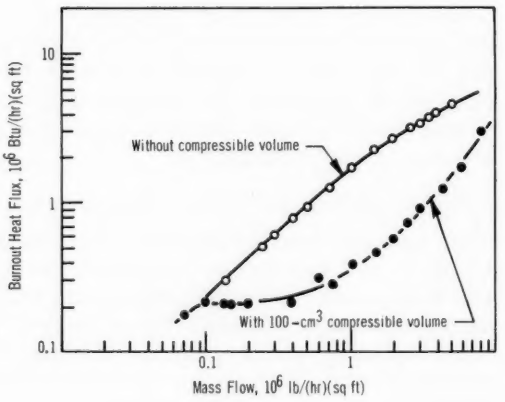

Fig. 2 Effect of compressible volume on burnout;[1] tube diameter was 0.096 in., L/D was 100, upstream pressure was 300 psia, and exit pressure (in downstream plenum, actually) was 15 psia.

BURNOUT HEAT FLUX AND FLOW STABILITY

Two sets of data from Ref. 1 are pertinent to this discussion. The first set surveyed the effects on burnout heat flux and flow stability of a compressible volume located between the flow restriction and the inlet of the test section. The sketch of test apparatus in Fig. 1 indicates that there was no pressure tap at the test-section exit and thus that the local pressure at burnout was taken equal to the downstream plenum pressure. Figure 2 shows Ref. 1 data for the cases where a compressible volume was and was not present. In Fig. 3 I have replotted their data points in terms of exit or burnout quality on the basis of atmospheric exit pressure, as they had assumed, and also have drawn curves showing the maximum possible flows that are predicted by two theoretical critical-flow models[9] on the basis of these same qualities. These predictions have been shown to bracket most existing adiabatic two-phase critical-flow data;[10] in fact, a recent burnout investigation (Fig. 4), in which critical flow occurred and was correctly accounted for, verifies the validity of these models for diabatic (nonadiabatic) flows as well.[11]

On the basis of assumed atmospheric-pressure conditions at the exit, the stiff-system (no compressible volume upstream) data of Fig. 3 would indicate that the flow was supercritical (supersonic). However, because supersonic flows actually are not obtained in constant-diameter ducts, the exit would have to be pressurized to satisfy critical-flow conditions. Thus the downstream pressure as measured in Ref. 1 cannot be the true exit pressure or local pressure at the point of burnout. In the case where there is a compressible volume upstream of the test section, flow oscillations initially present are most likely amplified, which leads to unstable burnout either before or at the initiation of choking as illustrated in

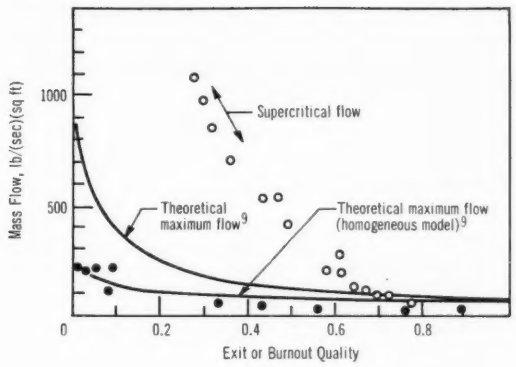

Fig. 3 Experimental data points of Fig. 2, when replotted in terms of burnout quality based on atmosphere exit pressure, indicate that flows would have been greater than maximum predicted by two theoretical models.[9]

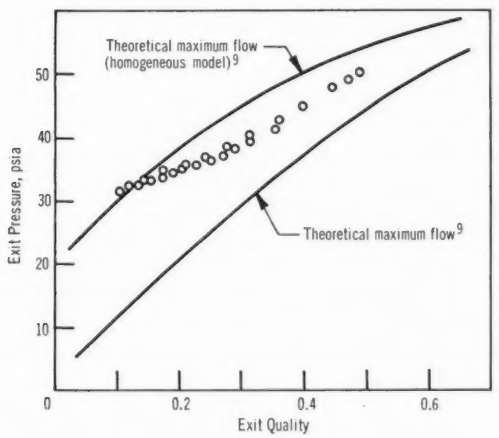

Fig. 4 Diabatic-flow data[11] between two curves predicted by models of Ref. 9 verify validity of theoretical models. Tube diameter was 0.094 in., L/D was 90, maximum flow was 10×10^6 lb/(hr)(sq ft), and downstream pressure was 30 psia.

Figs. 2 and 3. Even if small oscillations were not present, the first attempt at choking would lead to instability because flow will decrease when the compressible volume upstream compresses.

Figure 5 shows data points[1] taken at very high inlet-liquid velocities (without compressible volume upstream). However, when the data are considered in terms of maximum flow rates possible under atmospheric-pressure conditions, Mach numbers would have to have

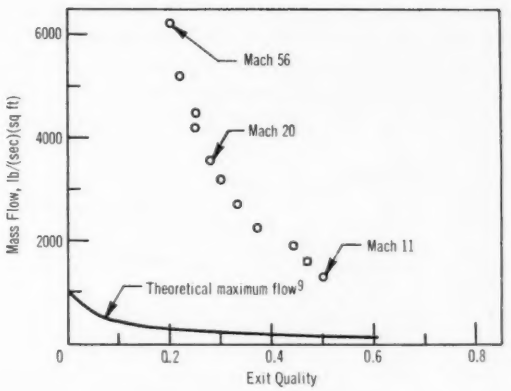

Fig. 5 Stiff-system flow-rate data points taken[1] at very high inlet-liquid velocities indicate highly supersonic flow if pressure at tube exit were truly atmospheric (as indicated by gauge in plenum downstream). Upstream pressure was 700 to 950 psia, tube diameter was 0.051 in., and L/D was 250.

been as high as 56 when exit quality was 0.2. Actually, critical-flow calculations indicate that the local conditions at the exit were probably more like "subcooled burnout" at approximately 200 psia.

PRESSURE DROP

The second set of pertinent data from Ref. 1 shows how varying the pressure drop across the flow restriction affects burnout heat flux and flow stability. Figure 6 shows an example of the data.[1] The same data, reevaluated to satisfy critical-flow conditions and exit pres-

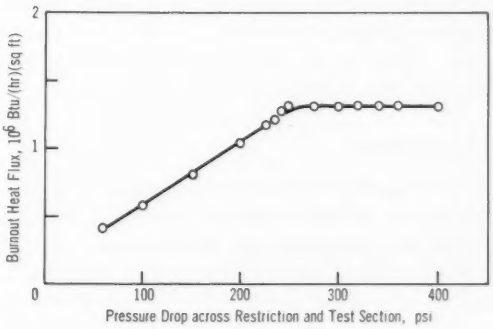

Fig. 6 Effect on burnout heat flux of varying pressure drop across flow restriction if the 15-psia downstream plenum pressure is assumed to be same as pressure at tube exit.[1] Tube diameter was 0.076 in., L/D was 250, and flow was 1.78×10^6 lb/(hr)(sq ft).

sures calculated by the two critical-flow models, are shown in Fig. 7. As can be seen, the choking may have caused up to 40% of the total pressure drop to occur downstream of the test section. This certainly explains the statement in Ref. 1, "The independence of the restriction pressure drop for a 3-to-1 variation in the test section pressure drop is unexplainable, inasmuch as both pressure drops occur upstream of the point of burnout." The primary result of choking is to decrease the available pressure drop across the restriction or valve upstream of the boiling channel; thus it is easier for small flow oscillations to affect the overall system parameters and to lead eventually to highly unstable flow conditions and burnout. Choking can therefore be interpreted as an indirect cause of flow instability. To accommodate sufficient throttling to maintain stable flow conditions despite the occurrence of choking, additional pumping head is required.

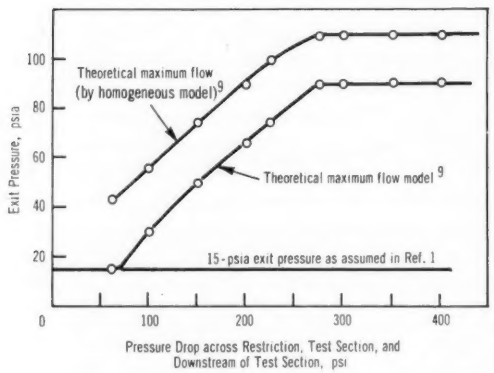

Fig. 7 Exit pressures were calculated by using two critical-flow models[9] to reevaluate Fig. 6 data[1] so that it satisfies critical-flow conditions.

Conclusions

This discussion shows that careful attention should be paid to a number of effects resulting from the presence of fluid compressibility within a flow system, especially choking or critical two-phase flow. Sufficient instrumentation should be installed in test loops to detect whether sonic effects are important. In any event, verified critical-flow models are available to check the experimental data for choking effects.

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Section

II

Power Reactor Technology

Components

Interpreting the French Liquid-Metal-Heated Steam-Generator Program

By Fred A. Smith

Development of reliable and economical sodium-heated steam generators is considered a major problem of sodium (fast reactor) technology in both the United States[1–3] and France.[4]

When the French work discussed in Ref. 4 is compared to our U. S. programs, we can appreciate the similarity of the objective (low cost power in fast liquid-metal-cooled reactors) and at the same time recognize that the French program is not redundant of our work; rather, it is different in design and thereby complements our efforts. We can sense that the problem of separating sodium and water may indeed have an economic solution.

Briefly, the French and U. S. liquid-metal steam-generator programs are similar in that both use once-through units at comparable temperature and pressure ranges. Further, both programs employ nonnuclear test facilities for detailed engineering evaluations of possible leakage. Essential differences are in the design approach—the French use a configuration of pipe headers and tubes rather than tubes and tube sheets in a shell design—and in the construction materials for both water and sodium sides. Another difference is in the size of test installations, 5 Mw(t) for French tests and 35 Mw(t) for U. S. tests.

U. S. Programs

Our own work consists in operating liquid-metal steam generators as components of sodium-cooled reactors and in testing other steam generators with nonnuclear heat sources.

OPERATING EXPERIENCE

The present U. S. nuclear plant operation of sodium-heated steam generators is primarily with EBR-II, Fermi, and Hallam. The salient design features and operating experience of these power-plant steam generators are described in a Geneva Conference paper.[5] Fermi and Hallam steam generators are of different design approaches but were fabricated by the same manufacturer. An article authored by the fabricator of these two plant units discusses the different design paths taken to reduce thermal stress and to minimize the possibility of tube failure.[6] The EBR-II steam-generator design and fabrication are Argonne National Laboratory efforts.[3]

The second part of the U. S. liquid-metal-heated steam-generator program is the testing of units up to 35-Mw(t) capacity in the nonnuclear gas-fired Sodium Components Test Installation (SCTI) located at Atomics International (AI).[7,8]

DESIGNS TO BE TESTED

Three industrially designed and fabricated U. S. steam generators are now completed or are being built for test in the SCTI. These designs, by AI, The Babcock & Wilcox Co. (B&W), and Alco Products, Inc./Baldwin-Lima-Hamilton Corp. (BLH), are shown schematically in Fig. 1.

The Alco/BLH unit, installed in the SCTI and being subjected to preoperational systems tests, uses composite tubes (type 316 stainless steel

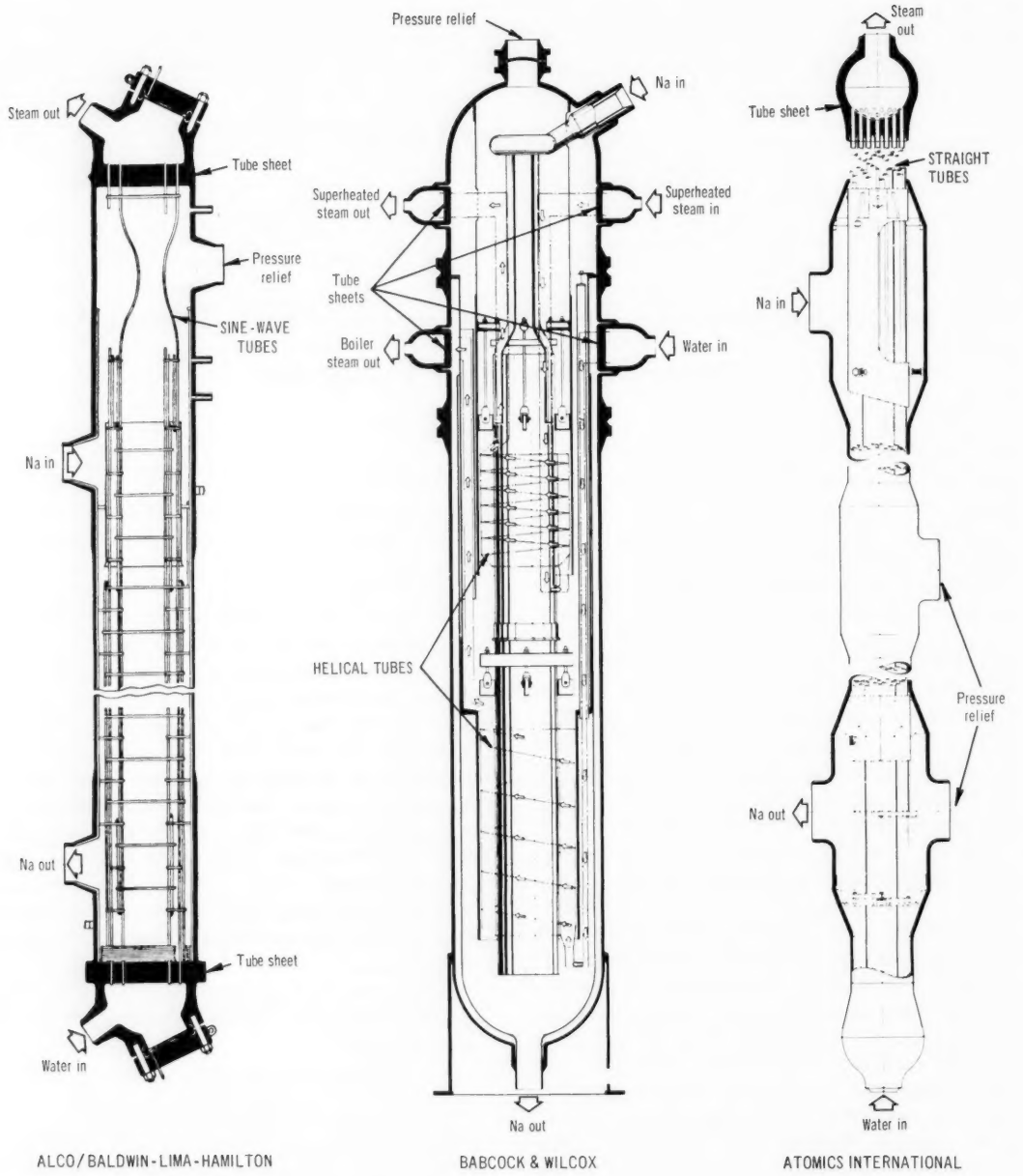

Fig. 1 Three U. S. steam generators[9-11] are being fabricated and will be tested in the SCTI. All are modifications of tube-and-shell designs.

on the outside and metallurgically bonded to Inconel) and incorporates a sine-wave bend of the tubes to accommodate thermal expansion.[9] The shell side of the tubes is exposed to sodium, and the inside diameter or tube side contains water/steam.

The B&W steam generator features helical coiled single-wall tubes for both the superheater and the evaporator.[10] These sections are vertically oriented in a single upright vessel. Evaporator tubing in the B&W unit is to be Croloy $2\frac{1}{4}$ because of its known compatibility

with saturated steam. Superheater tube material is type 316 stainless steel with good stress characteristics at high temperature. Both materials are considered compatible with sodium. A 25-Mw(t) unit of the design is being fabricated for test in the SCTI and is considered a prototype of a 1000-Mw(t) steam generator.

The AI design employs a modular tube-and-shell unit with only a small number of tubes per module.[11] There is no sine-wave bend, helical coil, or expansion joint to accommodate thermal expansion of the tubes; however, there can be a slight bowing of the tube bundle within the shell. The design is tailored to the use of Croloy 5 for the evaporator and 321-H for the superheater. The modular design features hemispherically forged heads, which, for large total power requirements (i.e., a large number of modules), might be economically produced.

All U. S. units now scheduled for nonnuclear testing are vertical once-through units employing modifications of a tube and tube sheet in a shell design and have sodium on the shell side. With the highlights of U. S. work in mind, let us see how the French design differs from, yet complements, our work.

The French Steam Generator

The French design, also a once-through unit, is shown schematically in Fig. 2 and consists

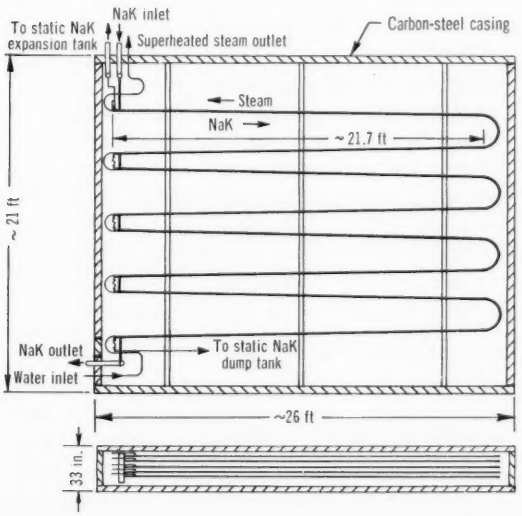

Fig. 2 Design configuration of the French liquid-metal steam generator is unique.[12] The unit is made up of large pipes that are headers for the heat-transfer tubes shown in Fig. 3.

of large pipes that are headers for the heat-transfer tubes. Each heat-transfer tube is a duplex annular configuration (Fig. 3) with liquid metal on the shell side and steam/water on the tube side. The two primary heat-transfer fluids are separated by an annulus of static NaK.

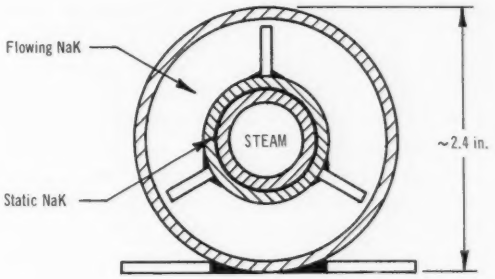

Fig. 3 In the heat-transfer tube, the two primary heat-transfer fluids are separated by an annulus of static NaK.[12]

Therefore the configuration of this steam generator differs considerably from the previously described U. S. designs.

The French unit is of 5 Mw(t) to match a 5-Mw(t) nonnuclear test facility. This test facility is discussed in detail in a new report[12] containing a plant description, flow diagrams, photographs of auxiliary-loop-equipment items, control features, and instrumentation. The heat-transfer area of the steam generator consists of the hairpin-shaped concentric-tube arrangements connected to appropriate pipe headers. This rather unique tube-and-header arrangement possesses several features of merit.

CAPABLE OF EXTRAPOLATION

Since pipe headers and bent tubes are the essentials of the design, the concept lends itself to extrapolation by simply adding tubes to the common pipe headers.

FAVORABLE WELD DESIGN

In the French unit the tubes are welded to relatively thin pipe headers, as compared to the welding of tubes to thick tube sheets in the more conventional tube and tube-sheet U. S. designs. The end result is that the French design should minimize thermal-stress problems that result from the use of sodium and the application of thin-walled tubes to very thick tube sheets. This point is treated in

considerable detail in Ref. 13, which discusses trends in the design of a variety of sodium components, including steam generators.

ACCESSIBILITY FOR MAINTENANCE

The French unit has only one concentric-tube configuration for each heat-transfer pass. The entire unit is contained in a simple accessible sheet-metal enclosure. It appears that this concept offers greater accessibility to each of the tube-to-header welds than is possible in a tube and tube-sheet design.

Maintenance and repair may well be required for any very large liquid-metal-heated steam generators. Operation of even a 5-Mw(t) French unit will be of considerable interest to determine just how easily maintenance is accomplished if required; i.e., we will learn more of importance to sodium-heated steam-generator development if this unit will just leak a little!

POSSIBLE CAPITAL-COST REDUCTION

The seeming simplicity of the French design can eliminate heavy thick forgings and the precision drilling required for tube and tube-sheet designs. Therefore precision drilling and any difficult welding associated with heavy tube sheets of conventional design are traded in the French concept for the welding of tubes to pipe headers. The possibility of cost reduction is related to the high cost for precision drilling compared to the low cost of pipe and conventional welding of pipe to pipe. The French unit requires a larger volume to accommodate a given capacity. The cost of space to house the unit must be weighed against the possible savings in maintenance. Pipe for heat transfer and for headers can always be relatively inexpensive compared to the cost of a high degree of dimensional precision required for the more conventional tube sheets.

HIGH DEGREE OF QUALITY CONTROL

The accessibility of the tube-to-pipe-header weld in the French design simplifies both visual inspection and X-ray inspection of weld quality. In conventional tube and tube-sheet design sodium-heated steam generators, tube-to-tube spacing is so small that it is extremely difficult to determine the quality of a weld after it is completed.

POSSIBILITY OF FIELD ERECTION

Very large steam generators of the French type can contain prefabricated components and be field erected. In the United States, designers of liquid-metal steam generators have taken advantage of the high heat-transfer capability of sodium (as compared to a hot gas) and have constructed steam generators in single vessels in shops; this has been the practice since the first sodium steam generators were considered and shop fabricated for submarine propulsion. These units, in general, were not accessible for repairs, and replacement was difficult.

It is not obvious that field erection of these generators implies an overall cost penalty, especially since large conventional boilers are field erected. Indeed, if the increased cost of relatively inexpensive building space for a French type unit does yield the ability to conduct maintenance, the overall capital cost plus maintenance cost of the French type may reduce the total cost of power. To emphasize the above point, B&W, in their classic reference book,[14] state a number of rules for a "perfect boiler" of the conventional type. One of these rules is: "All parts readily accessible for maintenance and repair. This point is of greatest importance as regards safety and economy."

In a justifiable concern to optimize the heat-transfer surface area and volume of sodium-heated steam generators, have we in the United States overlooked one of the cardinal rules of a "perfect boiler"? Not entirely. A 1959 USAEC-sponsored design study (Ref. 15) has proposed a sodium-heated steam generator similar in concept to the 5-Mw(t) unit actually built by the French; further, the AI modular steam generator offers the advantages of a highly subdivided unit. Therefore, for the reasons discussed above, it will be most interesting to compare operating costs of present U. S. designs with the operating cost of the French design.

NaK—Water Reactions

In addition to the test of steam generators in a nonnuclear test facility, the French reference discusses a separate engineering study of NaK–water reactions. For example, a typical tube wall is perforated using an electric-arc device that produces a 0.08-in hole. Depending on the selected leak location, water or steam reacts with the NaK. For the geometries tested to date (at 1800 psig and 1000°F), no

NaK-side pressure above 350 psig has been measured.

These tests complement (not duplicate) the small-leak sodium-water reaction tests currently being performed by Atomic Power Development Associates, Inc. (APDA). This work is discussed in detail in a recent APDA report (Ref. 16). For the purpose of this review, it is sufficient to point out that both the French and the U. S. programs include consideration of the nature and consequences of a leak between liquid metal and water/steam.

Conclusions

We should consider the French work as important and complementary to our own U. S. programs. After all, our mutual goal is to develop and confirm a set of rules for a "perfect liquid-metal-heated steam generator," which, by B&W definition (Ref. 14) for conventional units, implies that one must learn how to live with a possible leak!

Operation of units in the United States and abroad which have engineering-design differences should result in different operating experiences. These collective experiences should tend to reduce the cost of power from future sodium-cooled fast reactors.

We should keep in mind the similar diversity of design concepts, the improvements in structural materials, and the reduced cost of power which have all evolved since the pioneering efforts of Thomas Savory, James Watt, and many others.

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Section

III

Power Reactor Technology

Operating Experience

Design Foresight Can Simplify Reactor Modification

By Frederick H. Martens

That major repairs or modifications probably will be necessary during the life of most reactors is one conclusion that can be based on the number and scope of reactor experiences related at a meeting of the Reactor Operations Division of the American Nuclear Society.[1–5] Although related reactor-control and -hardware problems were discussed previously,[6,7] a review of the substantial modifications made to the Sodium Reactor Experiment (SRE), the Big Rock Point reactor, and the Heavy Water Components Test Reactor will suggest to reactor designers features that can simplify the task of making future modifications or, perhaps, eliminate the need for many of them. Of course, the techniques used also can be adopted by other operators faced with major repairs or changes.

Although each reactor to be modified presents unique problems, two general lessons seem apparent: (1) reactor-coolant inlet and outlet pipes should be sized to handle the maximum power that the reactor vessel can accommodate, and (2) reactor-vessel internals should be designed to be removable. In many instances reactor plants whose initial piping was adequate for higher powers have had their objectives and reactor powers increased accordingly. In other plants, where the incentive was strong enough, the original inlet and outlet piping has been replaced with larger pipes; but, because of the presence of radiation and shielding, these replacements usually present a great challenge to the ingenuity and cost considerably in money, effort, radiation exposure, and time. If it is physically or economically impractical to replace undersize components, reactor output will be limited, perhaps sufficiently to make the plant obsolete.

Reactor-vessel internals should be designed to be easily removable. This feature can allow replacement of malfunctioning core components, reduction of radiation exposure to workers when in-core changes or improvements must be made, and adaptation of the facility to new purposes when its original function has been served.

We recognize that it may be expensive (or virtually impossible) to incorporate these features. However, at the same time during the life of any experimental reactor, these features will probably be needed.

Sodium Reactor Experiment

The Sodium Reactor Experiment has been modified; the AEC is evaluating its usefulness for operation as a test facility for advanced sodium technology. The manner in which the changes were made illustrates useful methods for reducing the background radiation to which workers are exposed.[1]

The modifications involved replacing both the radioactive primary sodium heat-transfer system and the nonradioactive secondary sodium system with new sodium systems that will permit the reactor to generate higher power, higher neutron flux, and sodium temperatures of 1200° F at the reactor outlet.

Fuel-cladding failures in 1959 released fission products to the reactor vessel and primary sodium system. Thus long-life radioactivity remained in the system. In addition, cutting into

the system presented problems of contaminating the sodium with atmospheric impurities as well as contaminating the atmosphere with the sodium, part of which was radioactive. Of course it was necessary to guard against a sodium fire.

REMOVE HOT SPOTS FIRST

Fission-product concentrations were located by radiation surveys. High-concentration areas were removed first according to a precise schedule that minimized personnel exposure, yet ensured work continuity. After a radiation survey determined the hottest component in an area, the crew immediately cleaned or removed that component. Decontamination or removal of one hot spot initiated another survey. Removal of the entire heat-transfer systems took seven weeks, during which the average exposure per man was 1 rem; one man received the maximum exposure of 1.8 rem, which is below the maximum allowable for seven weeks.

SODIUM-FIRE PRECAUTIONS

Precautions were taken to prevent fires and to protect personnel and to contain any potential fire. A fireman was in attendance at all times. Protective equipment and clothing were provided. For containment of any radioactivity that might be released if a fire were to occur, the reactor room was kept under negative pressure relative to the surrounding area.

Fire-preventive precautions included: (1) suspension of work when relative humidity exceeded 50%, (2) cutting of pipe by procedures that caused minimum friction heat, and (3) immediate sealing of pipe cuts with plastic caps and tape. Because of the extreme care taken, there were no fires.

CLEANING

Components removed from the systems were cleaned and disposed of either as salvageable material or as radioactive waste. The cleaning involved the following: (1) both radioactive and nonradioactive components were immersed in a static 300°F oil bath where about 90% of the sodium melted and settled to the bottom of the oil tank; (2) secondary-system (nonradioactive) components were steam-cleaned and water-rinsed to remove the residual 10% of sodium, then were stored as normal salvageable material; (3) primary-system (radioactive) components were soaked in butyl cellosolve for safe removal of residual sodium. Components for disposal were wrapped in plastic and boxed as normal, dry, and noncompressible hot waste.

CORE-COMPONENT REPLACEMENT

The changes to SRE may make it useful for fuels and materials tests.[2] However, the procedures for replacing core components had to be streamlined to permit more rapid changes of the fuel and moderator. Actual replacement of a core in the reactor vessel takes nearly 40 days. This time, which includes fuel unloading, replacement of the moderator, and refueling, will probably be reduced with future development.

Two shielded machines have usually been used in the core change: one for the fuel elements and the other for the moderator elements. Removal of the plugs from three eccentric ports in the top shield gives access to the reactor vessel for moderator replacement.[8] Briefly, the sequence consists of installation of a slide valve over an access port, removal of the access plug, removal and replacement of all moderator elements beneath the port, encapsulation and shipment of each element to interim storage.

Critical reviews of these operations have eliminated much of the conservatism initially required in the development of an experimental nuclear plant. For example, the requirement for a shielded cask when access plugs are being removed has been eliminated. Although lower ends of the plugs may be contaminated to the extent of 0.5 r/hr, most of this contamination consists of beta emitters, so the operator is protected adequately if he stands a short distance from the plug. A plastic boot adequately contains the reactor atmosphere and surface contamination of the plug. Even if the boot were ruptured, the low cover-gas activity would prevent airborne activity from exceeding the maximum allowable concentration. Elimination of the need to move heavy casks has probably improved personnel safety. Thus, in addition to lowering costs and decreasing reactor downtime, these critical reviews have contributed to safety.

Improvements made in many other operations required for moderator replacement include: (1) rigging for top-shield rotation, (2) simple tool designs for use beneath the shield, (3)

elimination of part of the moderator encapsulator, and (4) construction of a permanent access platform. These improvements have made core-replacement routine, and future modifications will further reduce core-replacement time by making the moderator handling machine interchangeable with the moderator transport cask.

Big Rock Point Reactor

It first became necessary to modify internals of the Big Rock Point boiling-water reactor[3] when bolts that fasten fuel channels to channel supports loosened and dislodged. This failure resulted from excessive vibration caused by inadequate distribution of coolant entering the vessel (Fig. 1).

Part of the repair work was done by divers (Fig. 2) wearing dry suits that were supplied air from the plant instrumentation system. Dry suits, rather than the more familiar wet suits used by scuba divers, were used to prevent contact with contaminated reactor water. Divers previously had successfully repaired reactor components on the Gas-Cooled Reactor Experiment;[9] they repaired two broken cooling-water lines and tightened studs that had loosened under thermal cycling. In both cases, barriers prevented the divers from coming too close to radioactive components.

VIBRATION-INDUCED FAILURE

After examination of the failed channels and bolts revealed that vibration had caused the failures, an investigation was made to determine the extent and source of the trouble. Accelerometers and strain gauges were attached to channels in the reconstituted core; results pinpointed faulty coolant-inlet-flow distribution as the cause of the problem. Further evaluation of the test data resulted in the installation of a baffle that more equally distributes the inlet coolant flow. While the design and construction of the baffle were proceeding, the core support plate was removed from the vessel, modified, and replaced with the new parts.

PART REPLACEMENT

Two tasks were found to be necessary: (1) removal and modification of the core support plate for the new baffle and reinstallation of both the modified plate and the new baffle and (2) modification of the 72 fuel channels to prevent the bolts from loosening and separating the channels and support-tube assembly. To make the modifications to the core support plate and baffles, divers worked in the water-filled vessel

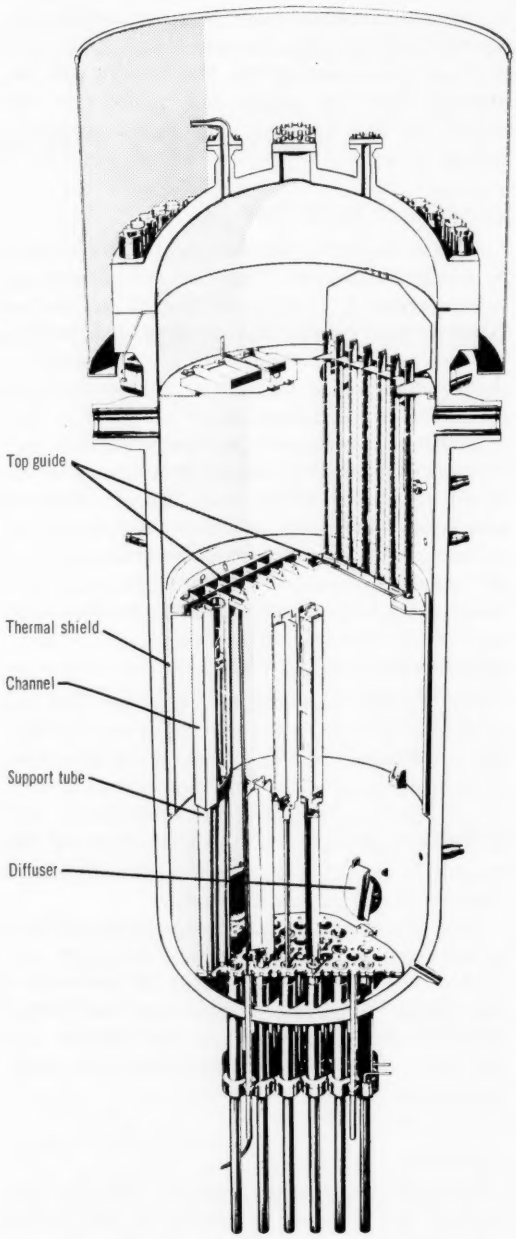

Fig. 1 The Big Rock Point reactor vessel and internals have been modified to correct inadequate flow distribution and resultant vibration, which loosened or fractured various bolts that connect the fuel channels to channel supports.[4]

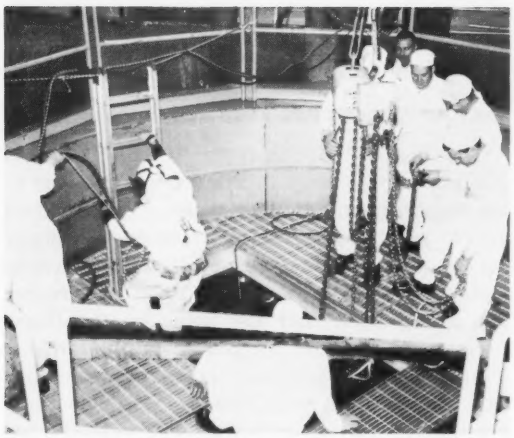

Fig. 2 Diver entering reactor vessel to work on core support plate.[4]

during each of the phases of removal and reinstallation. These tasks were accomplished without overexposure to radiation, even though measurements at some points inside the vessel indicated dose rates up to 120 r/hr.

The channel-modification work required construction and use of a shielded cave with lead-glass windows and remote tools for examination, cleaning, welding, and inspection. This work was finished with no radiation exposure problems, even though the channel section on which the work was being performed measured up to 1250 r/hr.

This Big Rock Point work established three distinct points: (1) the feasibility of working on highly irradiated parts by using temporary shielding and exercising considerable care and work control, (2) diving procedures can be used inside power-reactor vessels, and (3) satisfactory results can be obtained with normal vibration-measurement equipment underwater in very high radiation fields (estimated to be several million roentgens per hour).

IN-VESSEL MACHINING

Recent modifications to the thermal-shield support (as a result of fractured holddown studs) demonstrate the feasibility of relatively heavy but precise machining operations for in-vessel repairs of irradiated reactor internals.[4] These operations, done 30 ft underwater with the aid of television, included milling, drilling, reaming, tapping, and welding, and were to tolerances as tight as 0.001 in.

Visual and metallurgical examination of thermal shield support guide pins and failed studs revealed that fretting had opened clearances from a nominal 10 mils to about $\frac{1}{8}$ in. These openings allowed thermal-shield movement that ultimately caused thread-root fatigue failure of the studs. The problem was solved by installation of six support stilts to rigidly connect the thermal shield to the vessel support; the stilts are flexible enough to accommodate differential expansion between the 304 stainless-steel thermal shield and the carbon-steel reactor vessel. However, stilt installation required precise machine work.

The thermal-shield bracket on each stilt had to be removed by milling through two 6- by 6-in. stainless-steel gussets. Two vertical holes (1.37 in. in diameter) were drilled through each original vessel support pad and new stilt support foot. Then four horizontal holes (1.37 in. in diameter) were drilled and tapped through the 1-in.-thick stilt and into the 1.5-in.-thick thermal shield. Bolts were installed in the horizontal holes and studs in the vertical holes; then two 1.5-in. holes were drilled and reamed to a 0- to 1-mil press-fit through each stilt and thermal shield, and dowel pins were inserted. Finally, all studs and bolts were torqued to 700 ft-lb, and all nuts and keepers were seal welded.

All machine work (except vertical hole drilling) was done with a Cincinnati-Bickford horizontal radial drill that had an air motor, special lubrication, and remote handling features for underwater use. Vertical holes were drilled with the use of a right-angle-drive air-powered portable drill motor with a special clamp-on jig. Underwater television proved invaluable for monitoring milling operations, aligning tools, and making detailed inspections.

Machining chips and other extraneous materials were collected by a chip catcher directly under the immediate work area, and by a 200-mesh nylon net that covered the entire bottom of the vessel and vessel wall.

Actual stilt installation took about nine weeks. However, almost 24 weeks were required for the entire operation, including: (1) initial investigation, (2) selection of proper modification method, (3) selection, testing, and fabrication of necessary jigs, fixtures, and tools, (4) vessel cleaning after work was completed, (5) final inspection and instrumented flow tests of the new

stilt installation, and (6) complete reconstitution of reactor internals and fuel reloading.

Heavy Water Components Test Reactor

When it became necessary to modify a baffle within the Heavy Water Components Test Reactor (HWCTR) vessel, it was decided to leave the core in place and, after removal of virtually all the vessel internals above the core so as to gain access to the baffle, to provide local shielding for the men who would work within the vessel.[5]

REASON FOR MODIFICATION

It became necessary to replace the original four support legs, which mounted the gas-stilling baffle to the top of the water-flow-distribution baffle, when it was discovered that four of the eight $^5/_8$-in. bolts in the support legs had broken.[10,11] Metallurgical examination showed the failure was caused by low-ductility rupture, which probably was caused by a combination of overstressing during installation and thermal stresses during operation. Thus it was decided to install a stronger baffle support structure.

The gas-stilling baffle originally was installed in the neck of the HWCTR vessel to prevent gas entrainment in the core heavy-water coolant.[12]

RADIATION CONTROL

Removal of the vessel internals above the horizontal shield took about three weeks. The shield, which is a 19-in.-thick layer of 50% D_2O and 50% steel balls, was left covered by 2 ft of D_2O. Radioactive contamination on the vessel wall and components, and tritium evolution from the D_2O moderator, caused additional problems. However, air-supplied plastic suits and thorough ventilation of the work area kept tritium uptake by personnel to a minimum; fresh-air ventilation was controlled to ensure that water vapor did not significantly downgrade the D_2O. Lead shielding, hung in strips around the vessel wall and applied to the work basket that was lowered into the reactor, reduced the radiation peak from 3.5 r/hr to 100 mr/hr or less.

The 30 separate entries into the vessel involved an average gamma exposure of 60 mr during an average stay of 1 hr; the peak exposure rate was 120 mr/hr. Bioassay measurements of tritium exposure added to the external gamma doses but did not raise exposures beyond 300 mr/week.

SIMILAR EBWR EXPERIENCE

The method of repairing the HWCTR was similar to that used in major modifications to the Experimental Boiling Water Reactor some years previously.[13,14] Work inside the upper section of the EBWR vessel was necessary to install two large (6 in. and 10 in.) feedwater and steam lines, and also some smaller lines, to uprate the reactor for 100-Mw(t) operation.

After the core and most internals in the upper part of the vessel were removed, the remaining components and vessel walls were washed down to eliminate as much radioactive contamination as possible. A shield platform was lowered into the vessel for workmen to stand on, local side shielding was employed, and the water level in the vessel was kept as high above the core as possible. Radioactivity in the work area was found to rise when the reactor water was agitated; the intensity would fall off again as the radioactive particles in the water settled again to the bottom of the vessel.

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Section

IV

Power Reactor Technology

Physics

Thermal and Epithermal Spectrum Measurements

By George S. Stanford

As a general survey of recent developments in techniques and the results of actual spectrum measurements, this article extends previous consideration of neutron-thermalization experiments[1] and relies on background presented there, as well as on previous reviews of the entire field of neutron thermalization[2] and experimental techniques.[3] It discusses the applications and limitations of differential and integral methods for measuring neutron spectra. For differential methods it refers to recent work with pulsed accelerators, mechanical choppers, and crystal spectrometers. It surveys integral techniques (mainly activation) and discusses available probe materials, various perturbations, and corrections required; integral measurements in both well-moderated spectra and in hardened Maxwellian spectra are discussed. The more elaborate "semidifferential" measurements and related mathematical techniques are mentioned, as are proposals for improving the reliability and usefulness of foil-activation methods for measuring neutron spectra. This article excludes discussion of measurements of related quantities, such as cross sections, the scattering law, or diffusion parameters. Fast-neutron spectra (above approximately 1 kev) are also not considered.

After discussing the value of being able to predict accurately the neutron-energy spectrum in a reactor design, we will consider recent developments in the techniques for measuring spectra in various experimental configurations and ways in which the data can be interpreted and applied for prediction purposes.

Spectrum Prediction

For design of a new reactor, it is necessary to predict the energy spectrum of the neutrons so that we can predict the critical mass, the amount of fuel required. The more accurately the spectrum can be predicted, the more precisely the critical configuration can be calculated and the smaller need be the overdesign to allow for uncertainties. Thus precise knowledge of the spectrum can lead to savings in capital costs and perhaps in fuel inventory for particular reactor designs. The ability to predict spectra accurately can be of even greater importance when the predicted performances of different types of reactors are being compared.

At present the construction of many large reactors is preceded by costly critical experiments because various quantities must be known more accurately than they can be predicted. Some of the uncertainties are caused by insufficient ability to predict the neutron spectrum.

Most of the spectrum-measurement programs under way are intended mainly to test the adequacy of the scattering "kernels"[1] on which calculated spectra are based. Therein lies the importance of spectrum measurements. Poole[4] assesses the current situation as: "The measurement of stationary spectra in poisoned moderators and of time-dependent spectra in pure moderators has enabled scattering kernels to be put on a firm basis, at least as far as the isotropic part is concerned. . . . Work on the understanding of spatially dependent spectra is well under way but still not complete."

MEASUREMENT TECHNIQUES

Neutron-spectrum data can be obtained by two distinct types of measurements, usually referred to as differential measurements and integral measurements.

Differential Techniques. A true measurement of a spectrum can be obtained only by a differential measurement, wherein a beam of neutrons from the experimental assembly of interest passes into a neutron spectrometer, which sorts the neutrons according to energy. Such sorting can be accomplished either by Bragg reflection from a crystal or by pulsing the neutron beam and measuring the time it takes the neutrons to drift a known distance. The pulses can be produced by using a rotating chopper, by pulsing a subcritical assembly with neutrons from a pulsed accelerator, or by actually using a pulsed reactor.

Two contrasting types of neutron spectra are shown in Figs. 1 and 2. The spectra in Fig. 1 are typical of the case where the neutrons are fairly

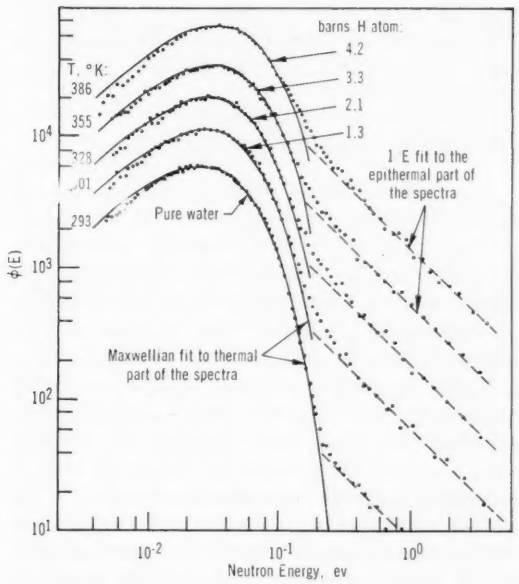

Fig. 1 Well-moderated neutron spectra in boric acid solutions for a 30-cm cube.[172]

well moderated and where there is no resonance absorption. On the other hand, in Fig. 2 the spectrum is not well moderated and the influence of the broad ^{239}Pu resonance at 0.3 ev and of the narrow ^{240}Pu resonance at 1.1 ev can be seen. (Note that the ordinate is $\phi(E)$ in Fig. 1, but $E\phi(E)$ in Fig. 2; thus a $1/E$ spectrum has a slope of $-45°$ in Fig. 1 but would appear as a horizontal line in Fig. 2.) Knowledge of the spectral shape [in the region] of the region of the 0.3-ev plutonium resonance is very important for calculating plutonium reactors.

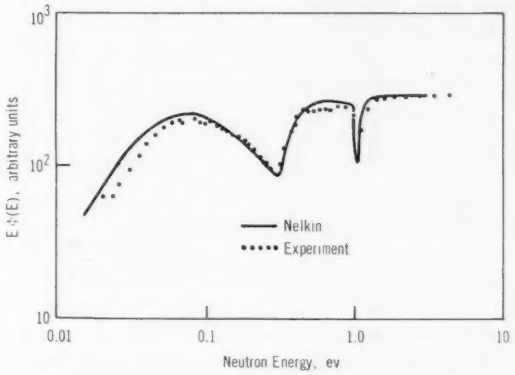

Fig. 2 Measured and calculated neutron spectra in a bare aqueous plutonium reactor[6] *where H : Pu = 131.*

For the differential technique the beam must be extracted through a hole that penetrates the assembly to the desired location. One must be confident that the neutron spectrum emerging from the tube is representative of the unperturbed spectrum at the desired location.[5] Representative beams can be extracted[6-8] if the flux gradients at the point of measurement (i.e., at the inner end of the beam tube) are not too steep and if the beam tube does not pass through regions of widely differing spectra.[4]

Integral Techniques. If it would be awkward to set up a neutron spectrometer or if the use of a spectrometer is precluded by low neutron intensity or by a requirement for high spatial resolution, it is nevertheless possible to obtain some information about the neutron spectrum by using integral techniques. Most integral measurements employ foils of materials with suitable neutron-activation characteristics. Several procedures are available, including comparison of the response of different materials, using sandwiches of foils, or surrounding some of the detectors with a filter such as cadmium. (For the purpose of this discussion, the "semidifferential" results sometimes obtained with reso-

nance detectors are included in the integral category because the energy resolution obtainable is so very much coarser than that provided by a spectrometer.)

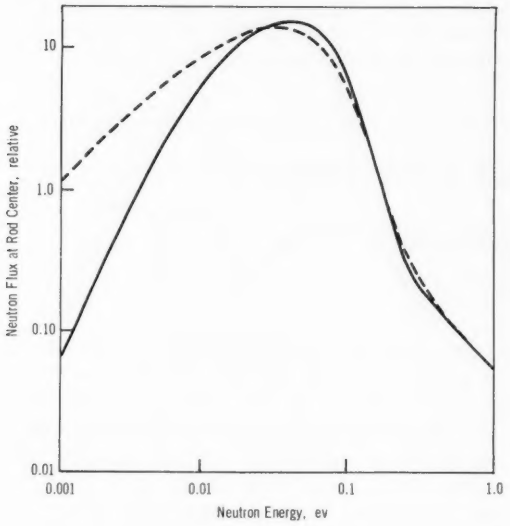

Fig. 3 Neutron spectra at center of an NRX fuel rod, calculated from the same activation data on the basis of two different spectral models.[9]

All these integral methods have a common requirement: to determine anything approaching a spectrum from the results, there must be a self-evident mathematical model of the spectrum, with one or more parameters to be determined from the measurements. The simplest such model in common use describes the spectrum as being Maxwellian in the thermal range, with a $1/E$ epithermal dependence. In this instance, only one parameter — the intensity ratio between the thermal and epithermal components — must be determined if the Maxwellian component is assumed to be in thermal equilibrium with the medium. If the cross section of the foil material is known as a function of energy, the intensity ratio can be determined from a single cadmium-ratio measurement.

Because different materials have differently shaped cross-section curves, the ability to compare the response of several detectors makes it possible to leave several parameters unspecified in the spectral model. Resonance detectors such as gold, indium, manganese, and cobalt can be used to determine departures from a $1/E$ epithermal shape. Indications of the shape in the thermal range can be found by using materials such as lutetium, europium, and plutonium, all of which have resonances below 1 ev.

Another perhaps fundamentally more significant use for integral measurements is as a check of detailed calculated spectra. Such a calculated theoretical spectrum can be used to predict response ratios for various detectors. If the observations agree with the predictions, the confidence in the calculational method is increased. However, because data available from integral measurements are inherently limited, it is possible that other spectral shapes could give equally good agreement with the data. Figure 3 provides an interesting illustration of this point.[9]

DIFFERENTIAL SPECTRUM MEASUREMENTS

Numerous measurements of differential neutron spectra have been reported, especially by active groups at Harwell, Rensselaer Polytechnic Institute (RPI), and the General Atomic Division of the General Dynamics Corp. Interesting background discussions of the various methods have been published.[4,10-12] The elaborate program of differential measurements under way at General Atomic has been discussed comprehensively.[12]

PULSED ACCELERATORS AND CHOPPERS

For time-of-flight type measurements of neutron energy, a neutron beam can be pulsed by accelerator or by a rotating mechanical chopper; similar results are obtainable with either system. The pulsed assembly provides greater neutron intensity, but it loses its resolving power if prolonged die-away times are involved, as with unpoisoned moderators or in assemblies with appreciable multiplication. However, for such cases a chopper can be used with the pulsed accelerator. Although such a combination has no intensity advantage over the reactor–chopper combination, it does permit the moderation process to be examined in more detail through the observation of the approach to an asymptotic spectrum with the passage of time.

This combination of chopper and pulsed accelerator has been used by groups at Harwell,[4,10,13-16] RPI,[17-24] General Atomic Division of General Dynamics Corp.,[25] and Karlsruhe.[26] Typically, the accelerator and chopper are synchronized to allow sampling of the spectrum at

preset times after neutron injection. At Harwell, however, this system has been abandoned for one in which the chopper opens at random times with respect to the accelerator pulse,[4,15] and the times are recorded on magnetic tape along with the time-of-flight data. Analysis of the resulting two-parameter data yields the time-dependent neutron spectrum, the average spectrum, or the asymptotic spectrum. Thus both methods give equivalent results. By permitting the moderation process to be observed as a function of time, the methods provide valuable checks of theory.

Some media in which die-away or asymptotic spectrum measurements have been made are graphite,[13,14] water,[21] polyethylene,[21] zirconium hydride,[26] beryllium,[20,27] heavy water,[15,19,22,27] ice at 77°K and 21°K,[26] and a variety of subcritical multiplying assemblies.[17,18,23,24]

Spatial Dependence. The technique of measuring spatially dependent asymptotic spectra provides "an excellent tool to study the extent of the region of space-energy separability, to determine the dependency of the extrapolation length on the buckling, and to investigate intensively the apparent difference between calculated and experimental (pulsed) values of the extrapolation length."[21] Such measurements have been used to determine diffusion-cooling coefficients[28] and energy-dependent extrapolation lengths.[8]

Steady-State Spectra. In measuring steady-state (i.e., time-intergrated) spectra with pulsed accelerators without benefit of a chopper, the finite die-away time of the neutrons in the assembly should be considered in analyzing the data. A theoretical basis for this correction has been presented.[29] Media in which stationary spectra have recently been measured with pulsed accelerators include graphite with copper and cadmium poisoning[30] and heavy water with boron and cadmium poisoning.[31] Further, investigations have been reported wherein the spectra in various media have been examined as functions of position of observation, angle of observation, size (buckling) of the assembly, or temperature of the medium. Leakage spectra also have been measured. Some of these experiments will now be considered.

Position of Observation. Position-dependent spectra have been measured in water and polyethylene,[21] in light and heavy water poisoned with $1/v$ and resonance absorbers,[12,32] in beryllium oxide poisoned with boron,[12] in a polyethylene cell, half of which was borated,[32,34] in uranium–water and uranium–monoisopropyl diphenyl lattices,[35] in beryllium poisoned with boron,[36] and in graphite near a temperature discontinuity.[37] The spectrum in water near a cadmium disk has been measured as a function of both position and angle of observation.[37]

Leakage Spectra and Angular Effect. Many spectra are measured near interfaces of various kinds. One reason for interest in spectra in such regions is that the spectrum within a homogeneous medium "is completely insensitive to the angular dependence of scattering and to that part of the scattering cross section involving no energy change of the scattered neutron, both of which are important in any spatially dependent problem."[37]

Leakage spectra from multiplying assemblies moderated by water or paraffin have been measured at RPI.[17,18] Angularly dependent leakage spectra from water, benzene, and Dowtherm A have been measured at Karlsruhe.[26] A similar Harwell measurement is reported for water, with the tentative conclusion that "a transport theory calculation which assumes isotropic scattering cannot predict the angular flux spectra at an absorption discontinuity in water."[31]

At General Atomic a strong flux gradient for the investigation of angular effects was created by placing a concentrated cadmium sulfate solution outside a thin tank containing dilute boric acid.[12]

Size Influence. Where the effects of leakage from an assembly are appreciable, the low-energy spectrum tends to show fine structure that correlates with the molecular or crystal structure of the medium. For example, the phenomenon of neutron trapping at the Bragg energies in beryllium has been observed during an RPI investigation of time-dependent spectra in beryllium assemblies of various sizes.[20] (Neutron trapping is evidenced by an increase in neutron density over the Maxwellian distribution near a Bragg coherent-scattering peak because a neutron of this energy tends, on scattering, to interact with a whole crystal; thus it loses no energy in the process.) Some of these results are shown in Figs. 4 and 5, where the buildup of neutrons at certain energies, relative to the Maxwellian, can be seen for various times after the neutron pulse is introduced to the assembly.

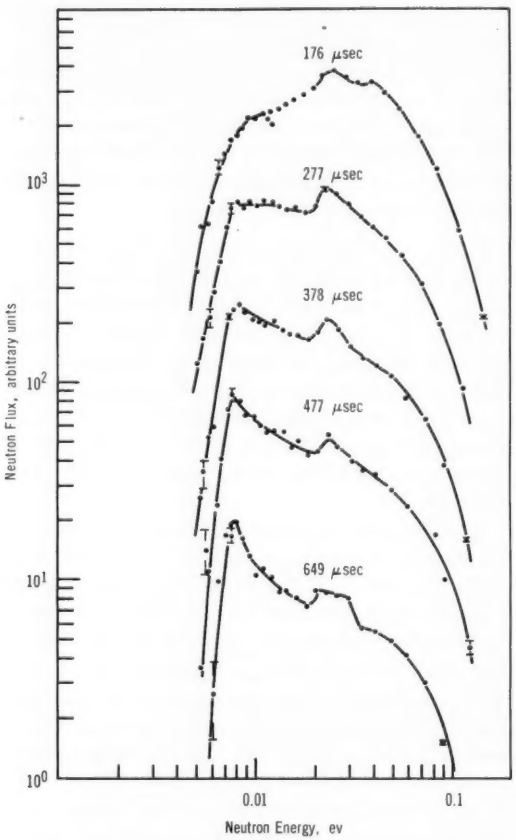

Fig. 4 Time-dependent spectra in beryllium metal[20] for array having $B^2 = 0.075\ cm^{-2}$. Normalization is arbitrary. As in Fig. 5, error bars are statistical only.

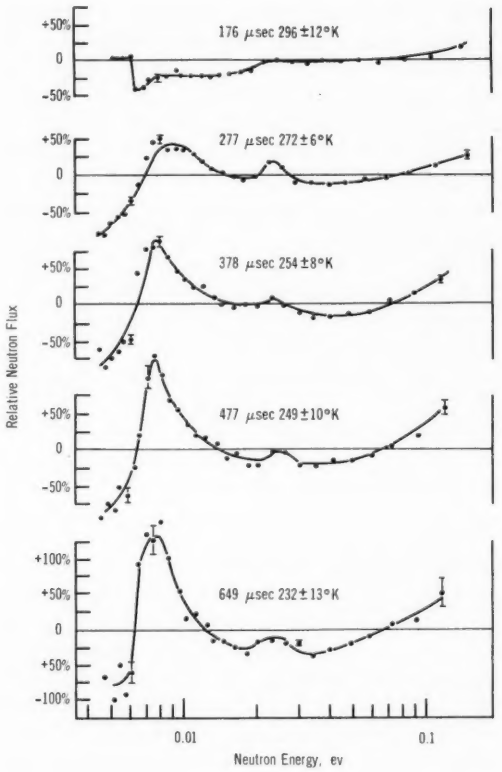

Fig. 5 Fractional deviation of beryllium neutron spectra from fitted Maxwellian distribution.[20] Medium temperature was 305°K; buckling was $0.075\ cm^{-2}$.

Temperature Effect. The influence of temperature on observed spectra has been investigated. For example, the spectra in ice at 77°K and at 21°K have been observed as a function of time.[26] At the higher temperature the neutron temperature approached an asymptotic value of approximately 75°K (indicating some diffusion cooling) with a relaxation time constant of approximately 75 μsec. For ice at the lower temperature, the relaxation time was approximately 800 μsec. At the other extreme, spectra have been measured in borated water at temperatures ranging up to 316°C in an effort to determine the importance of molecular binding in establishing the neutron spectrum in hot water.[34]

Spectra at elevated temperatures in borated graphite,[38] in samarium-poisoned graphite,[39] in zirconium hydride,[40,41] and in water poisoned with gadolinium or erbium[42] have also been measured. The spectra in zirconium hydride shifted markedly with temperature, in agreement with calculations "assuming the hydrogen atoms to be bound in an isotropic harmonic potential and allowing for the possibility of acoustical vibrations in the zirconium hydride lattice."[41]

REACTORS AND CHOPPERS

A chopper can be used in combination with a reactor to measure the spectrum of a beam extracted either from the reactor or from some other assembly that has been fed with reactor neutrons.

At AB Atomenergi,[43] a fast chopper was installed in a vertical beam emerging from a central hole in the R1 reactor. Neutrons were reflected upward by a block of scatterer placed in the hole at the reactor center. The measurements

covered energies from thermal to 10 kev with scatterers of heavy water, graphite, bismuth, and lead. The lead scatterer produced the beam that best represented the reactor spectrum; graphite also was good but produced a dip at 5.2 ev because of a 20-ppm silver impurity. The bismuth also was found to have a silver impurity (10 ppm) and in addition gave peaks at 790 and 2290 ev because of scattering resonances. As shown in Figs. 6 and 7, the D_2O gave generally poor results for energies greater than thermal.

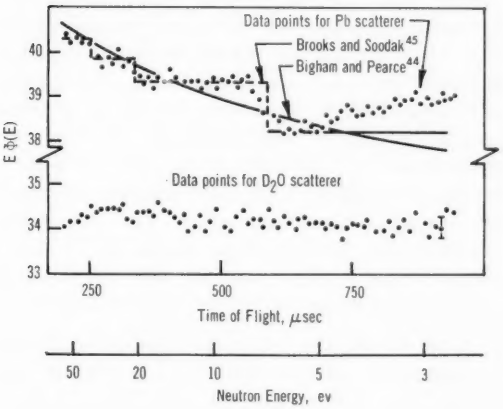

Fig. 6 Calculated and measured spectra for the region 2.6 to 65 ev; effects of the uranium resonances are apparent in data obtained with lead scatterer.[43]

In Figs. 6 and 7 the effects of the ^{238}U resonances at 6.7, 21, and 37 ev can be seen in the results for the lead scatterer as compared with the two theoretical calculations.[44,45] The deviation from $1/E$ (which would be a horizontal line) is clear. Note that the gold activation resonance at 4.9 ev, which frequently is used as a spectral indicator, occurs near the low point in the spectrum. Measurements of the thermal part of the spectrum indicate that the thermal neutrons are about 30°C hotter than the moderator. Spectra inside uranium tubes surrounded by D_2O have also been measured.[46,47]

Other chopper measurements of spectra in reactor lattices have been reported.[10,35,48-51] At Harwell the spectra in graphite lattices heated to various temperatures have been measured[10,50,51] and compared with detailed calculations. In connection with some of these measurements, a theoretical discussion of the

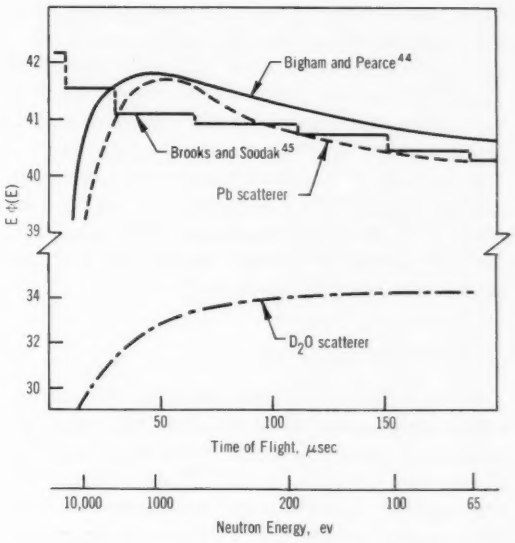

Fig. 7 Spectral data[43] *for the region 65 to 10,400 ev; curves continue from Fig. 6 and both sets of theoretical curves are normalized to lead at 400 μsec.*

analysis of chopper measurements contains bases for calculating such quantities as the emergent-beam spectrum, the spectrum actually observed, the geometrical weighting function, and the average cutoff function for a variety of rotor conditions—including "gray" rotors and off-center slits.[52] An extensive series of measurements has recorded: (1) spectra in water and graphite with various kinds and concentrations of poisoning, (2) spectra in graphite as a function of distance from a temperature discontinuity (a measurement of "rethermalization"), and (3) spectrum as a function of angle near a cadmium disk in water.[37]

Another rethermalization experiment was performed in Russia, where a neutron source was created in the middle of a container of water by irradiating a 4-cm-diameter lead sphere with a collimated beam of reactor neutrons that passed through a tube to the lead. The beam neutrons were at higher temperature than the water. With a chopper to measure the spectrum in the water as a function of distance from the source, the neutrons were found to be in thermal equilibrium with the water at distances greater than 2 cm from the lead sphere.[53]

At Knolls Atomic Power Laboratory, spectra have been measured in assemblies moderated by

water and by polyethylene: the assemblies were fed neutrons from the thermal column of a reactor.[54] Good agreement with calculations was observed for the polyethylene, but some discrepancies were noted for the important case of highly absorbing water-moderated lattices.

In contrast with some other reactor spectrum measurements,[55-58] a measurement in the fuel solution of the Forschungs-Reaktor Frankfurt has not shown a "bump" where the $1/E$ and Maxwellian components meet.[59]

DIFFRACTION SPECTROSCOPY

Several diffraction-spectroscopy type differential measurements of the slow-neutron components of reactor spectra have been reported during the last few years.[60-62] At Livermore a lead-crystal spectrometer was used for spectrum measurement; the report discusses design considerations and problems of measuring reactor spectra.[60] At the Naval Research Laboratory, a diffraction spectrometer with beryllium or germanium crystals was used in an experiment to optimize the thermal flux in a reactor beam.[61] The use of a NaCl crystal to measure a thermal-neutron spectrum also has been reported.[63]

SPECTRUM COMPILATIONS

Two compilations of thermal and near-thermal neutron spectra have appeared recently. One is a collection of spectra measured and calculated at Harwell.[64] The other[65] is a preliminary "Spectrum Book" reminiscent of early editions of the "Barn Book."[66] The loose-leaf arrangement of this ambitious undertaking will facilitate the addition and replacement of pages; it is intended that spectra measured both at General Atomic and elsewhere will be included. Theoretical calculations and the ratio of theory to experiment have been plotted along with the empirical data. Because it includes a careful description of the physical conditions under which each spectrum was measured, such a compilation should be useful for checking calculational methods. It also permits the construction of assemblies with known spectra so that activation type probe materials can be calibrated. Both these collections are to be incorporated in a worldwide compilation being assembled by the European–American Committee on Reactor Physics.

INTEGRAL SPECTRUM MEASUREMENTS

Although most integral spectrum work is done by activating foils (which will be the main topic of this subsection), two other methods are worth mentioning: reactor oscillation and neutron-beam transmission.

Reactor Oscillation. In this method the reactivity change is observed as each of a series of samples is inserted in and removed from a reactor. Because the oscillation method is not limited by the decay characteristics of activated sample material and because reactivity changes are directly measured, the technique sometimes is a more direct way to measure the quantity desired. However, the samples must be considerably larger than the activation probes (which increases self-shielding effects), the measurements take longer, and there are perturbations caused by the tubes used for inserting and removing the samples. The oscillation method has been used to obtain spectral indices[67,68] and to determine the effective neutron temperature inside heated graphite sleeves.[69]

Beam Attenuation. Where a neutron beam can be extracted, filters of various materials and thicknesses can be interposed and the attenuation observed. Something resembling a spectrum can then be calculated.[70-73] With this technique a temperature resolution of about 4°K was reported in an experiment that used boron filters to measure the spectrum along the vertical axis of a reactor.[73] However, most integral measurements are made with foils.

FOIL ACTIVATION

The remainder of this article will be concerned almost exclusively with measurements using activation of foils—the term "foil" here loosely meaning any small sample (even powder or liquid) used as an activation probe.

Detector Materials. Some of the foil materials that have been used in integral spectrum investigations are listed in Table 1 (in order of increasing energy of the principal resonance), along with pertinent physical constants. The "beta factor" and "gamma factor" are the number of countable beta and gamma rays emitted per disintegration as estimated from data in Ref. 74. By "countable" we arbitrarily mean gamma rays above about 50 kev and beta particles with transition energies greater than about 200 kev.

Table 1 DETECTOR CONSTANTS[8–15,17]

Detector	Isotopic abundance, %	Energy of principal resonance, ev	Half-life	Thermal-neutron cross section, barns: For element, $\sigma_{abs.}$	Thermal-neutron cross section, barns: For isotope, $\sigma_{act.}$	Resonance integral for isotope,* barns	Beta factor	Gamma factor
^{164}Dy	28.18		140 min	960	2800	377	1.00	0.14
^{50}Cr	4.31		27.7 days	3.1	15.9	7.11†	0.00	0.09
^{31}P	100		14.3 days	0.19	0.19	0.085†	1.00	0.00
^{176}Lu	2.59	0.142	6.72 days	108	2048.9	978	0.93	0.10
^{239}Pu	100.00	0.297		740.6	740.6‡	333‡	1.00	1.00
^{151}Eu	47.82	{0.327, 0.461}	9.3 hr	4300	1400	1200	0.74	0.30
^{191}Ir	37.30	{0.654, 5.36}	74 days	460	960	3500	0.95	2.00
^{193}Ir	62.70	1.303	19 hr	460	130	1370	0.34	0.60
^{115}In	95.72	1.457	53.99 min	194	150	1400§	1.00	2.10
^{185}Re	37.07	2.156	90 hr	85	120	1160	0.92	0.12
^{175}Lu	97.41	2.61	3.68 hr	108	13.26	476	1.00	0.10
^{23}Na	100.00	2.85	14.968 hr	0.53	0.524	0.28	1.00	2.00
^{165}Ho	100.00	3.92	27.2 hr	65	65	860	1.00	0.25
^{187}Re	62.93	4.41	17 hr	85	69	305	1.00	0.10
^{197}Au	100.00	4.906	2.697 days	98.8	98.8	470§	1.00	1.00
^{109}Ag	48.65	5.20	252.5 days	63	3.2	1160	0.95	2.5
^{152}Sm	26.63	8.01	47 hr	5800	140	3162	1.00	0.80
^{186}W	28.41	18.8	24 hr	19	34	420	1.00	0.30
^{123}Sb	42.75	21.6	60 days	5.5	2.5	138	1.00	1.80
^{74}Se	0.87	27	120 days	12	26	11.6†	0.00	1.80
^{75}As	100.00	47	26.8 hr	4.3	4.3	36.8	1.00	0.60
^{139}La	99.91	73.5	40.2 hr	8.9	8.9	13	1.00	2.35
^{198}Pt	7.21	95	30 min	10	4	53	1.00	1.00
^{59}Co	100.00	132	5.24 years	38	38	67	1.00	2.00
^{55}Mn	100.00	337	2.575 hr	13.22	13.22	11.8	1.00	1.40
^{98}Mo	23.78	480	66 hr	2.6	0.51	9.9	1.00	1.20
^{68}Zn (*a*)	18.57	530	13.8 hr	1.1	0.099	0.17	1.05	0.95
^{68}Zn (*b*)	18.57	530	55 min	1.1	1.0	0.45†	1.00	0.00
^{63}Cu	69.09	580	12.838 hr	3.8	4.5	4.4	0.57	1.20
^{64}Zn	48.89	2750	246.4 days	1.1	0.47	0.67	0.56	1.56

*Resonance integrals include $1/v$ part.
†$1/v$ contribution only.
‡Fission cross sections.
§Corrected for resonance self-shielding, for indium and gold thicknesses of 15 and 50 mg/cm^2.

An attempt was made to use current data in preparing Table 1, but a listing there does not constitute a recommendation. Most of the data came from Refs. 74 to 81, but the resonance integral of ^{198}Pt is from Ref. 82. The half-lives and thermal isotopic activation cross sections of ^{115}In, ^{23}Na, ^{75}As, ^{59}Co, ^{55}Mn, and ^{63}Cu, and the resonance integrals for ^{115}In, ^{59}Co, ^{55}Mn, and ^{63}Cu are values recommended by the Euratom Dosimetry Group.[76]

Irradiation and Counting. The choice of foil materials for a given use depends partly on the available neutron flux. Fluxes required to activate the various detectors have been calculated from the data in Table 1, with certain assumptions, and the results are given in Table 2. The assumptions, which pertain to the irradiation and counting conditions, are:

- Irradiation time is 30 min at constant flux.
- The resonance (epicadmium) flux per unit energy is proportional to $1/E$.
- Counting is started 1 hr after termination of the irradiation, except for some of the activities of lutetium, rhenium, iridium, samarium, antimony, selenium, molybdenum, and zinc, where there is an interfering shorter-lived activity that necessitates a longer wait.
- The total time available for counting is 4 hr.
- If a single foil were to be counted continuously for the full 4 hr, 10^7 counts would be recorded.
- Separated isotopes are not used.
- The foil weight is the lesser of (1) 100 mg of the natural element or (2) the amount that would cause a 1-cm^2 foil to have a thermal $l\Sigma_{abs.}$ (for the natural element) of 0.015.
- Flux depression and self-shielding are ignored, except for the resonances in gold and indium; this leads to lower-limit resonance fluxes for materials such as silver, cobalt, and manganese, where resonance self-shielding can be appreciable.

Table 2 FLUXES AND COUNTING RATES

Detector	$t\Sigma_{abs.}$ (thermal)	Foil weight, mg	Count rate needed to total 10^7 counts in 4 hr, counts/sec			Neutron fluxes, neutrons/(cm^2)(sec)		Relative fluxes*	
			Initial	After 1 hr	After 3 hr	Thermal, beta gamma	Resonance, beta gamma	Thermal, beta gamma	Resonance, beta gamma
^{164}Dy	0.01500	4.25	1187	882	487	9.39×10^6	6.97×10^7	0.11	3.7
						6.70×10^7	4.98×10^8	0.75	26
^{50}Cr	0.00373	100.00	696	695	694	–	–	–	–
						1.80×10^{11}	4.02×10^{11}	2,000	21,000
^{31}P	0.00037	100.00	697	696	693	1.87×10^{10}	4.19×10^{10}	210	2,200
						–	–	–	–
^{176}Lu	0.01500	40.5	700	697	691	7.97×10^8	1.67×10^9	8.9	89
						7.42×10^9	1.55×10^{10}	83	830
^{239}Pu	0.01500	8.04	867	773	613	1.16×10^7	2.57×10^7	0.13	1.4
						1.16×10^7	2.57×10^7	0.13	1.4
^{151}Eu	0.01500	0.87	803	745	642	1.36×10^8	1.59×10^8	1.5	8.5
						3.38×10^8	3.94×10^8	3.8	21
^{191}Ir	0.01500	10.4	695	695	694	3.72×10^9	1.02×10^9	42	54
						1.77×10^9	4.85×10^8	20	26
^{193}Ir	0.01500	10.4	746	720	669	4.74×10^8	4.50×10^7	5.3	2.4
						2.68×10^8	2.55×10^7	3.0	1.4
^{115}In	0.01500	14.76	2243	1038	222	1.36×10^7	1.46×10^6	0.15	0.08
						6.50×10^6	6.70×10^5	0.07	0.04
^{185}Re	0.01500	54.21	705	700	689	9.41×10^8	9.73×10^7	11	5.2
						7.21×10^9	7.46×10^8	80	40
^{175}Lu	0.01500	40.5	989	819	562	7.40×10^7	2.06×10^6	0.83	0.11
						7.40×10^8	2.06×10^7	8.3	1.1
^{23}Na	0.00137	100.00	761	726	662	2.54×10^8	4.75×10^8	2.8	25
						1.27×10^8	2.37×10^8	1.4	13
^{165}Ho	0.01500	63.22	730	712	677	3.94×10^7	2.98×10^6	0.4	0.16
						1.58×10^8	1.19×10^7	1.8	0.63
^{187}Re	0.01500	54.79	753	723	666	5.50×10^7	1.24×10^7	0.61	0.66
						5.50×10^8	1.24×10^8	6.1	6.6
^{197}Au	0.01500	49.66	709	702	687	8.95×10^7	1.88×10^7	1.00	1.00
						8.95×10^7	1.88×10^7	1.00	1.00
^{109}Ag	0.01500	43.09	695	695	694	3.45×10^{11}	9.52×10^8	3,900	51
						1.31×10^{11}	3.62×10^8	1,500	19
^{152}Sm	0.01500	0.65	715	705	684	1.06×10^{10}	4.74×10^8	120	25
						1.33×10^{10}	5.93×10^8	150	31
^{186}W	0.00615	100	735	714	674	1.69×10^8	1.37×10^7	1.9	0.73
						5.63×10^8	4.55×10^7	6.3	2.4
^{123}Sb	0.00269	100	695	695	694	5.68×10^{10}	1.03×10^9	630	55
						3.16×10^{10}	5.72×10^8	350	30
^{74}Se	0.00977	100	695	695	694	–	–	–	–
						1.74×10^{11}	3.90×10^{11}	1,900	21,000
^{75}As	0.00345	100	731	712	676	1.69×10^8	1.98×10^7	1.9	1.0
						2.82×10^8	3.29×10^7	3.1	1.8
^{139}La	0.00386	100	719	706	682	2.21×10^8	1.51×10^8	2.5	8.0
						9.41×10^7	6.44×10^7	1.1	3.4
^{198}Pt	0.00304	100	3866	966	60	3.53×10^9	2.66×10^8	39	14
						3.53×10^9	2.66×10^8	39	14
^{59}Co	0.01500	38.67	694	694	694	6.14×10^{10}	3.48×10^{10}	690	1,800
						3.07×10^{10}	1.74×10^{10}	340	920
^{55}Mn	0.01448	100	1134	866	506	8.14×10^6	9.12×10^6	0.09	0.48
						5.82×10^6	6.52×10^6	0.06	0.35
^{98}Mo	0.00160	100	709	702	687	1.88×10^{10}	9.69×10^8	210	52
						1.57×10^{10}	8.08×10^8	180	43
^{68}Zn (*a*)	0.00097	100	767	729	659	4.25×10^{10}	2.47×10^{10}	470	1,300
						4.69×10^{10}	2.73×10^{10}	520	1,400
^{68}Zn (*b*)	0.00097	100	2208	1036	228	9.08×10^8	2.03×10^9	10	110
						–	–	–	–
^{63}Cu	0.00363	100	772	732	657	1.80×10^8	1.85×10^8	2.0	9.8
						8.58×10^7	8.77×10^7	0.96	4.7
^{64}Zn	0.00104	100	695	695	694	1.04×10^{12}	7.27×10^{11}	12,000	39,000
						3.72×10^{11}	2.61×10^{11}	4,200	14,000

*Relative fluxes are with respect to 1-mil gold.

● For the "countable" betas and gammas, the overall detection efficiency of the counter, including geometry, is 10%.

Successful counting requires count rates low enough that dead-time and stability problems are not serious, yet high enough that background corrections are not excessive. The three columns labeled "Count rate needed..." in Table 2 will assist in planning experiments. They show the counting rates that would lead to an accumulation of 10^7 counts in 4 hr of continuous counting. For other counting conditions, Tables 1 and 2 provide a convenient starting point for estimating the relative or absolute neutron flux required to activate a given foil.

Self-Absorption. In counting activated foils, it is sometimes necessary to consider self-absorption by the foil of the radiation being detected, particularly when counting for absolute (rather than relative) activation, in making thickness corrections, or in extrapolating to zero thickness. In most gamma-ray counting, the effect of self-absorption is small, but in beta counting it is usually appreciable. Investigations and discussions of beta-ray self-absorption are available.[3,83,84]

CORRECTIONS IN FOIL IRRADIATIONS

In general, when foils are used to investigate a neutron flux, that flux will be somewhat perturbed by the presence of the foils. Two types of disturbance must be considered, self-shielding and outer flux depression.

Self-Shielding. The outer layers of an absorbing foil reduce the neutron flux irradiating the interior of the foil, and cause the flux, averaged through the volume of the foil, to be less than the flux present at its surface. This effect, called self-shielding, is particularly noticeable at the energies of large resonances such as occur in gold and indium; it also is important at thermal energies if the thermal absorption is appreciable. Self-shielding is a function of the foil and of the neutron spectra but not of the medium in which the foil is embedded for the irradiation.

Outer Flux Depression. A function of both foil and medium, outer flux depression occurs when the flux at the surface of the foil is less than it would be if the foil were nonabsorbing. This effect is important mainly at thermal energies, because in the slowing-down part of the spectrum a neutron usually cannot pass twice through the foil without losing energy between passes—so that the presence of a resonance in the foil usually cannot reduce the resonance-energy flux impinging on it.

Perturbation Studies. There have been several important investigations of theoretical and experimental aspects of the problem of foil-induced flux perturbations. Although this article is concerned primarily with experimental results, several important theoretical papers are noteworthy.[85-89] From Ref. 89 the thermal-region perturbations can be written.[90]

$$\frac{\bar{\phi}}{\phi_0} = \frac{[\frac{1}{2} - E_3(\tau)]/\tau}{1 + [\frac{1}{2} - E_3(\tau)]g}$$

where[90]
ϕ_0 = average flux at foil location before insertion of the foil
$\bar{\phi}$ = average flux after insertion
τ = macroscopic absorption cross section of the foil times its thickness
$E_3(\tau)$ = third-order exponential integral

The numerator gives the self-shielding of the foil, and the denominator gives the outer flux depression. The function g, which depends on both the foil and the medium, can be determined according to a method in Ref. 90.

Although the above equation was derived for isotropic flux, it is valid even for strong anisotropies[91] if the foil is at least several mean free paths from an absorbing object, such as a fuel rod.[92]

Discussions of the theoretical and experimental status of thermal-flux perturbations caused by absorbing foils have been published.[93-95] Good experimental agreement with theory has been reported for gold and indium foils in light-water media.[96,97] Measurements of thermal-neutron self-shielding in various materials have been described.[98] "Edge effects" on the activation of finite foils have been investigated.[94,99] A fictitious thickness $t^* = t/[1 + (t/R)]$, where R is the foil radius, has been proposed as a way to compensate for edge effects in calculations of the self-shielding of small foils.[100]

If liquid foils can be used, the solution composition sometimes can be adjusted to match the moderating ratio of the medium. When this is done, the thermal-neutron perturbations largely disappear.[101-103]

For the epithermal region a calculation and experimental results of resonance self-shielding for indium have been reported.[104] When accurate corrections are being made for small thickness variations in sets of foils, self-shielding, if it is appreciable, must be considered because the activation is not proportional to the thickness. The first derivative of the theoretical self-shielding curves can be used for this purpose.[105]

At Savannah River Laboratory cadmium ratios were measured as a function of foil thickness, for many of the more commonly used materials, and for two locations where the thermal-to-epithermal flux ratios were quite different.[106,107] The epithermal self-shielding correction factors were the same for both positions, indicating that these factors tend to be insensitive to the neutron spectrum. Thus these empirical results are of general utility.

For an inclusive discussion of the theory of probe-induced perturbations and of techniques for obtaining and interpreting foil-irradiation data, see Ref. 3.

Effects of Cadmium Covers. Because of its unique combination of properties, including a large near-thermal resonance, cadmium is used widely as a foil cover to separate thermal and epithermal activations. Of the many spectral indices that can be derived from foil irradiations, the "cadmium ratio" is the most widely used. Because it is the ratio of the activation of a cadmium-covered foil to that of a bare foil exposed in the same flux, it indicates the thermal-to-epithermal flux ratio.

Precise interpretation of a cadmium-ratio measurement, however, requires careful consideration of a number of factors. For example, it is necessary to know the effective "cadmium-cutoff energy," (a point at about 0.5 ev) which is a function of (1) cadmium thickness, (2) foil material, (3) the neutron spectrum, and (4) the degree of flux isotropy. Computer calculations of cadmium-cutoff energies for a wide variety of conditions have been reported.[108-114] Effective cutoff energies for foil covers of boron, gadolinium, and samarium have also been calculated.[110,112] In using the results of these calculations it should be recognized that it is customary to define the "cutoff energy" in such a way that (1) it is artificially reduced by the activation due to subcadmium neutrons that penetrate the cadmium cover and (2) it also is affected by attenuation by the cadmium of epicadmium neutrons at energies corresponding to resonances in the foil.

Alternately, it is possible to define the cutoff energy as the effective value for a $1/v$ foil with the activation caused by Maxwellian neutrons excluded. This exclusion allows separate handling of the effects caused by subcadmium transmission and by epicadmium absorption by the cadmium at foil resonances.[115] Determinations of a correction factor F_{Cd}, obtained by extrapolating to zero-thickness cadmium, have been reported.[115-119] Such an extrapolation involves an implicit change in cutoff energy. Because the various effects involve complex interactions, erroneous or duplicate cadmium-cover corrections may result if the exact meanings of the various published cutoff energies and correction factors are not observed.[115]

Measurements in Well-Moderated Spectra

In the use of foils to determine spectral information, the method selected and the treatment of data depend on the kind of spectrum being investigated. The simplest case is a well-moderated spectrum in a homogeneous medium where the spectrum is well represented by a Maxwellian thermal component, and an epithermal part where flux per unit energy is proportional to $1/E$, the two being connected by some sort of "joining function." Such spectra are shown in Fig. 1. For this example an effective cross section $\hat{\sigma}$ for an absorber can be represented by the Westcott formula $\hat{\sigma} = \sigma_{2200}(g + rs)$, where g and s are measures of the degree to which the thermal and epithermal components of the differential absorption cross section, $\sigma(E)$, differ from the $1/v$ form, and where r is indicative of the relative intensities of the thermal and epithermal portions of the spectrum.[120-122] Computed values of g and s for many materials are available.[122]

Discounting the possible uncertainty in the joining function,[122] such well-moderated spectra are completely specified by two parameters, such as the temperature, T, of the Maxwellian component and the Westcott r. With the Westcott formalism, either a cadmium-ratio measurement or a measurement of the relative activations of two materials with differing g or s can be used to determine the quantity $r\sqrt{T/T_0}$, where T_0 is 293.6°K.

Thermal-to-Epithermal Ratios. Many measurements of $r\sqrt{T/T_0}$, or equivalent quantities, have been reported, frequently in conjunction with determinations of the neutron temperature T.[123-131] To monitor the average value of $r\sqrt{T/T_0}$ over the lifetime of a high-power reactor core, ^{109}Ag and ^{59}Co have been recommended as a suitable pair of activants.[126]

Although the traditional spectral index has been the cadmium ratio, cases have arisen where the use of cadmium foil covers is either impractical (for long irradiations at high flux because of loss of the ^{113}Cd by neutron capture) or undesirable (because of flux perturbations or space limitations). Equivalent information can be obtained by comparing activations of different detectors.[81,132] However, because either absolute counting is required or the detector pair must be calibrated in a known spectrum, it seems inevitable that cadmium will continue to be widely used—probably often when it should not be.

Dysprosium Substitution Method. Straightforward cadmium-ratio measurements run into difficulties when a hard spectrum is combined with an activant where resonance integral is large relative to its thermal-activation cross section. In this case the cadmium ratio approaches unity, which causes problems because the cadmium ratio frequently appears in the expression (CdR-1). However, a means for measuring (CdR-1) directly has been devised.[133,134] The procedure has been called the "thermal-activation" or "dysprosium-substitution" method. It involves measuring the cadmium ratio for dysprosium (which has a relatively small resonance integral) in the unknown spectrum, along with a bare sample of the detector of interest, e.g., ^{238}U. To avoid the necessity for absolute counting, a simultaneous run is made in a highly thermal spectrum.

Neutron Dosimeters. In an inversion of the usual arrangement, a neutron dosimeter has been developed that consists of a cadmium foil between two thick gold foils; it provides both spectral and directional information.[135] After exposure, both gold foils are beta-counted on both sides to determine the front-to-back activation ratios. Because of resonance self-shielding, the gold surface facing the neutron source will have the highest activation (the cadmium shields the back foil from thermal neutrons). With the activities of the four gold surfaces, a set of simultaneous equations can be used to define the epithermal and thermal flux components entering each side of the detector.

Neutron-Temperature Measurements. For a well-thermalized spectrum this can be determined from the activation ratio of two materials whose thermal-activation cross sections have different shapes—i.e., for which the values of g differ. Typically, a detector with a low-lying resonance, such as ^{176}Lu or ^{239}Pu (see Table 1), is combined with a material with a $1/v$ thermal cross section, such as ^{175}Lu or ^{55}Mn. Such a method has been discussed in some detail.[3,80,136] Several other discussions and sets of measurements also have been published.[6,84,137-144]

In some cases foils have been replaced with small fission chambers,[6,145,146] but the principle is the same.

Activation techniques can be used conveniently to investigate the spatial variation in neutron temperature near boundaries. Measurements of temperature perturbations have been reported near absorbers,[147,148] in and near cylinders of a hot moderator immersed in a cool moderator,[125] and as a function of position in a reactor lattice.[130,149,150] Some differential measurements of neutron rethermalization have already been mentioned. Similar measurements of the space dependence of neutron temperature near a temperature discontinuity have also been made with foils.[127,151]

Reference 149 contains a comprehensive bibliography, and Ref. 152 discusses activation techniques for measuring Westcott type spectra.

MEASUREMENTS IN HARDENED MAXWELLIAN SPECTRA

When a medium has a fairly large macroscopic absorption cross section, the thermal component of the neutron spectrum can not be represented accurately by a Maxwellian.[153] A first-order allowance for deviations from the Maxwellian can be made by using more than one pair of thermal detectors, so that more than one thermal parameter can be determined.[154-157]

Some comparisons of the observed response of thermal detectors with the response calculated on the basis of measured spectra or theoretical models have been published. Activation ratios observed in reactor lattices demonstrate the necessity of including chemical bind-

ing effects in scattering kernels[158] and also for considering scattering anisotropy.[159]

EPITHERMAL SPECTRA: DEVIATIONS FROM 1/E

The epithermal component of a neutron spectrum frequently deviates significantly from $1/E$. If the deviation is caused by resonance absorption—for example, by ^{238}U or plutonium in a reactor lattice—the spectrum will have a fine structure that correlates with the resonance energies (Figs. 2 and 6).

Figure 8 shows a case of a medium with strong thermal leakage, which should be contrasted with Fig. 1. The epithermal spectra in Fig. 8 are smooth, but they deviate markedly from $1/E$, which would be a horizontal line. It also can be seen that the concept of a Maxwellian component is not applicable.

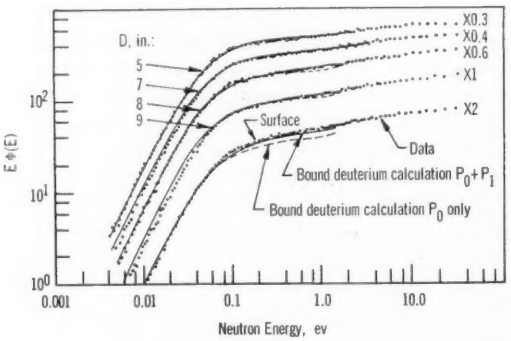

Fig. 8 Calculated and measured differential neutron spectra in leaky heavy-water assembly poisoned to 1.94 barns/atom of deuterium with 1/v absorber.[32]

To the extent that the epithermal spectrum is linear on a log–log plot, the flux per unit energy is proportional to $E^{-\alpha}$, where α would be unity for a $1/E$ spectrum. For such a case it is clearly possible to determine α when the epithermal activations of two or more detectors having different resonance energies in the epithermal region are known. Such a measurement has been reported,[160] and the effective resonance integral of thin gold as a function of α has been calculated.[161]

Several other investigations that used the same principle but expressed the deviations from $1/E$ in different ways have been reported,[142,162,163] as have cases [82,144,164] in which deviations from $1/E$ were sought but not found.

SEMIDIFFERENTIAL SPECTRA

As Table 1 shows, the principal resonances of the available activants cover a wide energy range. Thus, in principle, it should be possible to determine something approaching a differential spectrum by activating various materials.

With the notable exception of ^{164}Dy, which shows the effects of a negative-energy resonance, most absorption cross sections are inversely proportional to the velocity of the incoming neutrons for energies below the region of the lowest lying resonance. (In most cases the lowest resonance is also the principal resonance by a wide margin.)

Data Analysis. In analyzing the data from a set of such probes, it is customary to subtract from the observed activation the "$1/v$ contribution," which is done by making initial assumptions about the shape of the spectrum being measured; the result is the activation caused by neutrons at or near the principal resonance energy. (Higher resonances are also considered.) Knowledge of the activation cross sections and resonance integrals permits calculation of the neutron flux at the energies of the resonances. Because several different materials must be used, absolute counting (or comparison with calibrated standards) usually must be used. However, cadmium-ratio measurements sometimes can be used to normalize all activations to the thermal component.

Results. Several measurements of this general nature have been reported.[124,128,131,165] In a specific application, the ^{240}Pu resonance escape probability in a plutonium-fueled reactor has been investigated with iridium foils.[166] The ^{191}Ir resonance at 0.654 ev and the ^{193}Ir resonance at 1.303 ev bracket the 1.054-ev ^{240}Pu resonance, making it possible to estimate the relative neutron flux above and below that plutonium resonance.

Resonance Self-Shielding. Spectral information can be obtained from different thicknesses of the same foil material by making use of changes in resonance self-shielding.[167,168]

Resonance self-shielding also is used in a "sandwich method." When three layers of a resonance absorber are sandwiched, the resonance activation of the center foil will be reduced to a greater extent than the nonresonance component. The difference between the inner and outer activities indicates flux at the reso-

nance energy.[3,95,169] Epithermal-neutron currents have also been measured by such a sandwich.[170]

MEASURED VS. CALCULATED ACTIVATIONS

Foils have been activated in certain neutron spectra that either have been more precisely measured by differential means or have been calculated on the basis of theoretical models, permitting comparison of the expected activations and the measured values.

Integral measurements in aqueous plutonium reactors have been compared at Harwell with calculations based on the Nelkin[171] scattering model for water and also with time-of-flight data.[6] "In general, the trend is for the measured (reaction-rate) ratios to lie closer to the experimental time-of-flight predictions than to the Nelkin predictions."[6] Lutetium–copper activation ratios have been compared with measured thermal spectra and with calculations based on the Nelkin model.[172,173] Here again, the agreement with the measured spectra was somewhat better than with the calculated ones. Fair agreement was observed with activation ratios calculated on the basis of the Nelkin kernel.[158]

Activations of resonance detectors (and also threshold detectors for the fast spectra) have been compared with calculations and with chopper measurements.[174] "The absolute values calculated agree satisfactorily with those measured in well-defined geometries," [but there can be large disagreements in boundary regions.] Observed reaction rates have been compared with those calculated from a measured thermal spectrum, with the conclusion that, for a foil that is covered with cadmium, the activation[109] cannot be calculated to better than 5 to 10%. A computer code for calculating foil activations is available.[175]

MATHEMATICAL EXPANSION TECHNIQUES

Several mathematical methods for deducing a neutron spectrum from a set of activation measurements have been proposed. Some of them have been discussed primarily in the context of fast spectra but are also generally applicable to the thermal and epithermal regions. Earlier work[176–178] forms the basis for some of the recent efforts listed in the bibliography.[157,165,179–182] The general approach is exemplified by Lanning's formulation,[182] where the thermal spectrum $\phi(E)$ is represented by a finite series:

$$\phi(E) = W(E) \sum_{j=1}^{k} A_j \psi_j(E)$$

Here $W(E)$ is an approximate spectrum, such as a Maxwellian or Wigner–Wilkins type, and the $\psi_j(E)$ are suitable linearly independent functions. The coefficients A_j are to be determined from the activation of k foils.

Results can be perturbed by experimental errors or inaccurate cross-section data, especially if the affected foil has a cross-section shape similar to that of one of the other foil materials.[179] If, however, the number of parameters to be determined is less than the number of foil materials, this problem can be alleviated.[183]

Limitations. Because the choice of available foil materials is limited, spectral resolution is likewise limited. In all these methods it is necessary to make an initial assumption about the functional form of the spectrum. The final result will depend on this initial assumption to an extent that depends on the number of probe materials, the precision of the cross-section data, the accuracy of the experiment, and the number of parameters being determined. The type of ambiguity that can be inherent in a one-parameter measurement is shown in Fig. 3.

LIMITATIONS OF INTEGRAL METHODS

Although it is easy to irradiate and count a foil, the apparent ease of determining accurate spectral information from foil activations is entirely deceptive. An approximate thermal-to-epithermal flux ratio can be inferred from a simple cadmium-ratio measurement, but interpretation difficulties mount rapidly when additional spectral information is required.

Factors Affecting Data Correction. Among the perturbing effects that must be considered in correcting the data are: (1) thermal self-shielding, (2) outer flux depression, (3) resonance self-shielding, (4) edge effects, (5) absorption of epicadmium neutrons by cadmium covers, (6) transmission of subcadmium neutrons by cadmium covers, (7) outer flux depression caused by cadmium covers, (8) self-absorption in the counting process, (9) deadtime in the counting system, and (10) spectrum differences between the surface and the inside of a thick

foil. These effects apply to determination of relative counting rates. But if absolute disintegration rates are needed, as, for example, when activation ratios of different materials are to be determined, the counting system must also be calibrated for proper decay corrections for each material, perhaps by a simultaneous irradiation in a known spectrum. Careful attention usually must be paid to the times involved—reactor periods, irradiation time, and elapsed time before counting.

For many materials, accurate activation cross sections, as functions of energy, are not available and the experimental uncertainties in many measurements of resonance integrals are large. Thus even if the experimenter takes great care to get accurate activation data, his efforts can be negated by inaccurate or insufficient cross-section information.

Various Approaches to Data Correction. The literature in this field indicates a large diversity of approaches. Although it often is possible to make all relevant corrections, the procedure is so complex that the ordinary experimenter who just wants to know his spectrum has neither time nor patience to follow the entire process. Improvement will come when:

- More accurate and more complete cross-section data are obtained.
- Irradiation and data-handling procedures are standardized to a greater extent. Suggestions for standardization already have been published,[184] and a committee of the American Society of Testing and Materials is working on the problem.
- Catalogs listing activation rates are assembled for a wide variety of possible spectra and irradiation conditions. Then an unknown spectrum can be identified by irradiating a suitable set of activants and searching the catalog to find the spectra that most closely match the activation data. Such a search could be done by a computer that also could be programmed to interpolate between the listed spectra. Some work of this kind has been reported[185-187] consisting of calculated activation rates expected in a number of different spectra. Similar activation-rate calculations are being made for fast spectra.[188-191]

Calculated activation rates have two inherent drawbacks. In the first place, accurate differential cross sections are a prerequisite. Second, even with accurate cross-section data for infinitely thin (i.e., nonperturbing) foils, numerous perturbing influences usually have to be considered in making the calculations. Ideally, the kind of catalog mentioned above would contain activation rates that have actually been measured under standard conditions in spectra that have been carefully measured by time-of-flight or other differential means. Such a proposal already has been made.[152,192]

In general, it should not be necessary to do the activation work at the same place where the differential measurements were made, because spectrum compilations[64,65] could permit assembly of environments where the spectrum is well-known. Such a project would be a sizable undertaking, but spectral information then could be obtained from foils, with no reliance whatsoever on the accuracy of differential cross-section measurements.

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